RADIATION RESISTANT CONCRETE FOR APPLICATIONS IN NUCLEAR POWER AND RADIOACTIVE WASTE INDUSTRIES

by

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STATEMENT OF THESIS APPROVAL

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ABSTRACT

Elemental components of ordinary concrete contain a variety of metals and rare earth elements that are susceptible to neutron activation. This activation occurs by means of radiative capture, a neutron interaction that results in formation of radioisotopes such as Co-60, Eu-152, and Eu-154. Studies have shown that these three radioisotopes are responsible for the residual radioactivity found in nuclear power plant concrete reactor dome and shielding walls. Such concrete is classified as Low Level Radioactive Waste (LLRW) and Very Low Level Waste (VLLW) by International Atomic Energy Agency (IAEA) standards and requires disposal at appropriate disposal sites. There are only three such sites in the USA, and every nuclear power plant will produce at the time of decommissioning approximately 1,500 tonnes of activated concrete classified as LLRW and VLLW. "NAVA ALIGA" (ancient word for a "new stone") is a new concrete mixture developed mainly by research as presented in this thesis. The purpose of NAVA ALIGA is to satisfy IAEA clearance levels if used as a material for reactor dome, spent fuel pool, or radioactive waste canisters. NAVA ALIGA will never be activated above the IAEA clearance level after long-term exposure to neutron radiation when used as a material for reactor dome, spent fuel pool, and radioactive waste canisters. Components of NAVA ALIGA were identified using Instrumental Neutron Activation Analysis (INAA) and Inductively Coupled Plasma Mass Spectrometry (ISP-MS) to determine trace element composition. In addition, it was tested for compressive strength and permeability, important for nuclear infrastructure. The studied mixture had a high water to cement ratio of 0.56, which likely resulted in the high measured permeability, yet the mixture also showed a compressive strength greater than 6 000 psi after 28 days. In addition to this experimental analysis, which goal was to develop a standard approach to define the concrete mixtures in satisfying the IAEA radiation clearance levels, the NAVA ALIGA concrete was analyzed as to potentially be used together with depleted uranium. This study was purely computational (based on MCNP6 models) and was twofold: to find if this new concrete mix would enhance the radiation shielding properties when combined with depleted uranium and to find if this will be an effective and useful way of using the existing large quantities of disposed depleted uranium.

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CHAPTER 1

INTRODUCTION

1.1 Motivation

One of the greatest challenges facing the future development of nuclear power is that of the cost of developing new infrastructure. Between lengthy permitting processes, cost of construction, and the future cost of disposal it is not financially feasible to develop new infrastructure without government subsidies. One of the greatest expenses associated with nuclear energy infrastructure development is the need to front the cost of disposal of radioactive waste which can be as much as $2,800/tonne in the United States and even higher in other countries [1]. In a structural containment dome the first 100 cm is exposed to neutron irradiation creating as much as 1,500 tons of concrete that is classified as Low Level Radioactive Waste (LLRW) and Very Low Level Radioactive Waste (VLLW) [1]. Due to shielding as well as structural safety, power plant containment vessels require large amounts of concrete and reinforcement able to withstand, in some instances, aircraft impact [2]. As a result this creates a large amount of concrete that can be subjected to neutron irradiation and become classified LLRW and VLLW.

Within concrete there exists certain nuclides, which cause it to become classified as LLRW and VLLW after being exposed to neutron irradiation. A reduction of these nuclides, specifically Co-60, Eu-152, and Eu-154, would reduce the long-term residual radioactivity of concrete. The elimination of these target nuclides has never before been considered in US nuclear power plant design but has however been researched with some success in Japan [3, 4, 5, 6]. The International Atomic Energy Agency (IAEA) has defined maximum allowable concentration and activity limits for these radionuclides to be considered nonradioactive waste as outlined in Table 1.1 and a calculation of a clearance level (D/C) that must be less-than 1 is shown in Fig. 1.1. Fig. 1.2 shows that concrete activated from a 40-year neutron flux of 2.5×10^5 contains only three elements having activities above the IAEA clearance level. Due to the long half-lives compared to other elements, these three isotopes do not decay away during the cool down period between reactor shutdown and when decommissioning starts. This demonstrates that concrete containing cobalt and europium remains radioactive for several decades.

Concrete is also a common material used for biological shielding in hospitals, waste storage, and linear accelerators. Research efforts have been undertaken to improve the shielding capability of concrete such as at Jefferson Lab where plastic waste and boron were added to concrete to enhance its neutron shielding capabilities[7]. Concrete could be incorporated with depleted uranium (DU) to create a high-quality biological shielding to be used in hospitals and radioactive waste repositories. A computational analysis using MCNP6 will show if concrete would enhance the radiation shielding properties when combined with DU and determine if this is an effective and useful way of using large existing quantities of DU.

1.2 Thesis Objectives

The objectives of this research are as follows:

1. Investigate new concrete components identified as potentially having low-activation characteristics using Instrumental Neutron Activation Analysis (INAA) and Inductively Coupled Mass Spectrometry (ICP-MS).
2. Create a concrete mixture using new concrete components identified.
3. Test the strength and permeability of the new concrete mixture.
4. Investigate shielding capabilities of depleted uranium when combined with new concrete mix.

1.3 Organization of Thesis

Neutron physics is described in Chapter 2. In Chapter 3 an in depth description of INAA is given as well as protocol at the University of Utah. Chapter 4 describes ICP-MS, specifically the instrument used in this research. Chapter 5 describes concrete chemistry and the investigation process of cement and aggregate sources. In Chapter 6 the physical testing of candidate samples is described. The properties of DU and computer modeling of the shielding capabilities are presented in Chapter 7. The work is concluded with recommendations for future work in Chapter 8.

Table 1.1: IAEA clearance levels for target radionuclides [8]

Isotope	Concentration	Activity
Co-60	<2.0 ppm	<0.1 Bq/g
Eu-152	<0.1 ppm	<0.1 Bq/g
Eu-154	<0.1 ppm	<0.1 Bq/g

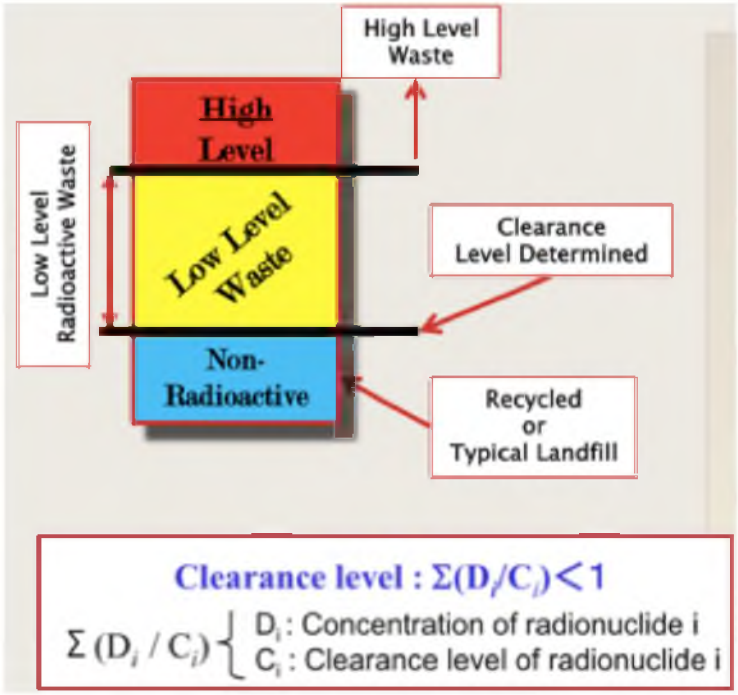

Fig. 1.1: Clearance level as defined by IAEA

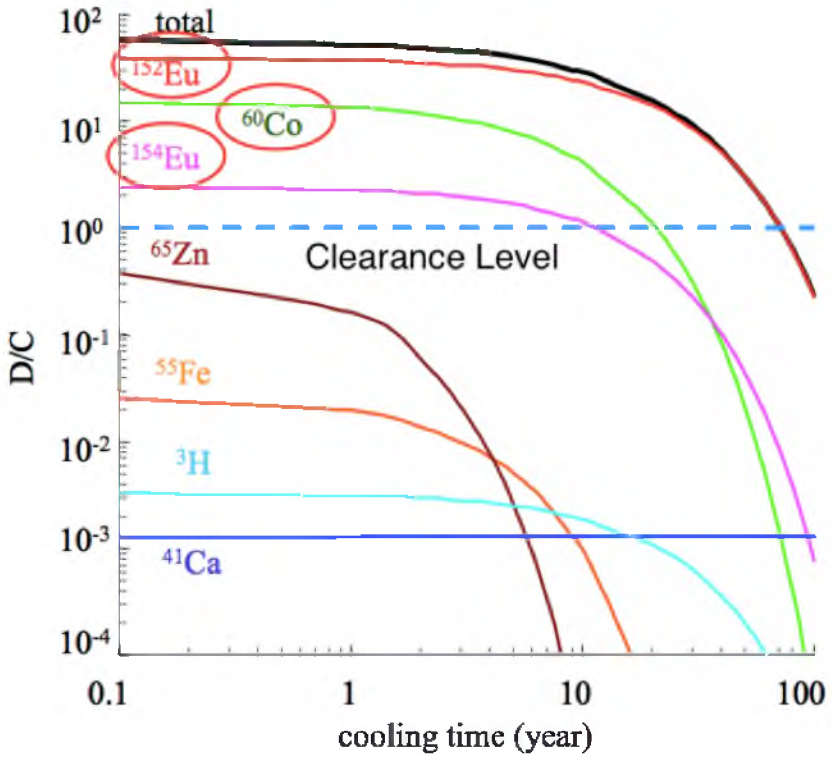

Fig. 1.2: Average D/C as defined in Fig. 1.1 of concrete used in nuclear power plant [9]

CHAPTER 2

NEUTRON PHYSICS

2.1 Introduction

Neutrons are electrically neutral and exist inside the nucleus, which is also composed of protons. Because the neutron is electrically neutral, it makes it easy for them to pass through the negatively charged electron cloud without an interaction and thus interact with the nucleus. Free neutrons are unstable with a half-life of approximately 14.6 minutes decaying into a proton, electron, and antineutrino. It is, however, much more likely for the neutron to interact with surrounding matter before it decays away [10].

The probability of a neutron to interact with a nucleus is defined by its microscopic cross-section (σ). The microscopic cross-section is measured in the unit of barns ($10^{-24} cm^2$) and is the probability of a given type of nuclear interaction, per nuclide, per unit time per unit flux [11]. There are a number of different types of nuclear interactions that can be divided into two categories: scattering interactions and absorption reactions, denoted σ_s and σ_a, respectively. The probability for any type of nuclear interaction to occur is defined as the total cross-section (σ_t).

2.2 Scattering Interactions

Scattering can be elastic (n,n) or inelastic scattering (n,n'). Elastic scattering is either resonance or potential elastic scattering. Resonance elastic scattering occurs when an incident neutron interacts with a nucleus and is absorbed to form a compound nucleus; a neutron is then re-emitted leaving the nucleus in its ground state [12]. Potential elastic scattering is the more common form of neutron elastic scattering and occurs when an incident neutron interacts only with the short-range nuclear forces of the nucleus. The neutron therefore is repelled before a compound nucleus can be formed. Because momentum and kinetic energy are conserved in elastic scattering, some amount is transferred to the target nucleus as the neutron is repelled. These cross-sections are nearly constant and are illustrated in Fig. 2.1 for Eu-151.

Inelastic scattering occurs in a two-step process. An incident neutron interacts with a nucleus placing it in one of its excited states. The nucleus then emits a neutron at an energy less than the

incident neutron, leaving the nucleus in a less excited state. The nucleus then emits a gamma-ray to bring it back to its original ground state. Because the nucleus is placed in one of its excited states, the neutron must be at or above some threshold energy for the interaction to take place. The first four inelastic scattering cross-sections with Eu-151 are shown in Fig. 2.2.

2.3 Absorption Reactions

Neutron absorption reactions are divided into different channels: radiative capture, fission, charged particle emission, and neutron emission [12].

Radiative capture or neutron capture (n,γ) is the most frequent nuclear reaction with neutrons [13]. This reaction is described with (2.1) where an incident neutron is absorbed, forming a compound nucleus, and a gamma-ray is then ejected.

$$ {}^{Z}_{A}X + {}^{1}_{0}n \rightarrow {}^{A+1}_{Z}X + \gamma \tag{2.1} $$

The likelihood of radiative capture is increased if the incident neutron is at thermal or resonance energy. The radiative capture cross-section of Eu-151 is shown in Fig. 2.3 and is usually divided into three regions: 1/v region, resonance region, and above resonance region. Neutron speed is proportional to the square root of energy, so the radiative capture cross-section varies as 1/v. The cross-section appears as a straight line with a slope of -1/2. The resonance region is observed by its distinct, narrow peaks where the cross-section can be several thousand barns. Neutrons at these specific energies have a higher likelihood of interaction than other energies. Above, the resonance region decreases continuously as neutron energy increases to very small cross-section values [14]. This is the most important type of neutron interaction for INAA because it is the gamma-ray emitted whose energy is measured and uniquely attributed to an isotope.

Charged particle emission is a type of neutron interaction in which an incident neutron is absorbed; an alpha particle or proton is emitted from the nucleus. An example of charged particle emission is the reaction of Boron-10 with a neutron shown in (2.2) and in Fig. 2.4.

$$ {}^{10}B + n \rightarrow {}^{10}B^{*} \rightarrow {}^{7}Li + \alpha \tag{2.2} $$

Fission is another type of neutron reaction and is best explained using the liquid drop model. If a single drop of liquid is envisioned and a force is applied to that drop, one of two things can happen. If very little force is applied, the drop will deform slightly and then return to its original shape. If enough force is applied, the liquid drop will start to deform in a similar manner as when little force is applied but will continue to deform until it splits into two drops. When a neutron interacts with a nucleus, if the energy is high enough the nucleus will continue to deform into a dumbbell shape at which point it cannot return back to its original shape and will then split into two.

If there is not enough energy to split the nucleus, then it will return to its original shape by ejecting a photon. It can therefore be concluded that fission and radiative capture are competing processes. If the incident neutron is above the critical energy, then fission will win; otherwise radiative capture will take place. The critical energy needed for fission to take place decreases as the value of Z^2/A increases, where Z is the atomic number and A is the mass number. For elements where the critical energy is less than the binding energy, such as U-235 and Pu-239, fission can occur with neutrons of all energies. These types of isotopes are called fissile, whereas elements such as U-238 and Th-232 are fissionable because the critical energy is greater than the binding energy [15].

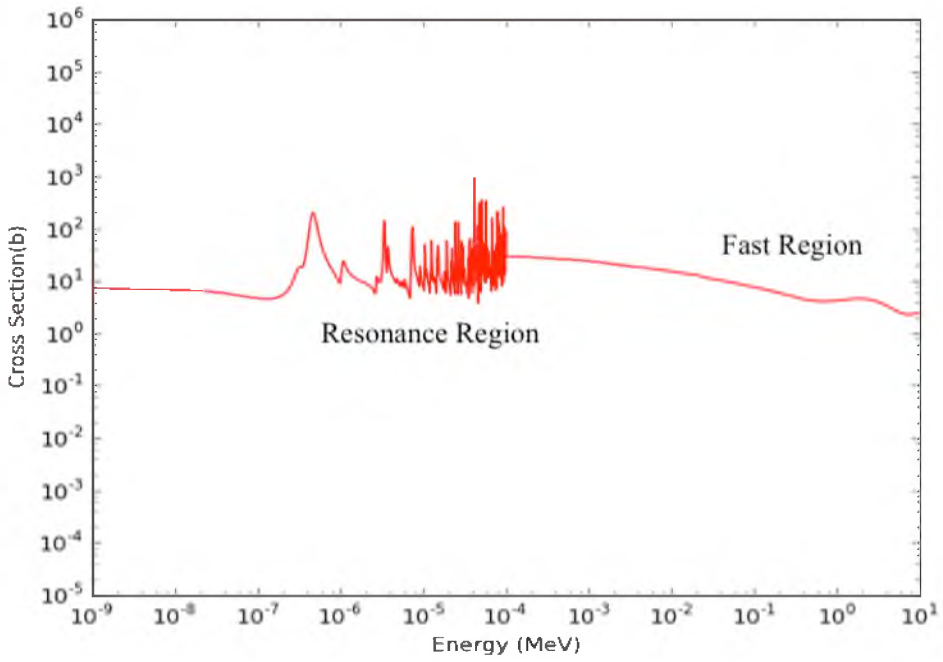

Fig. 2.1: Elastic scattering cross-section for Eu-151[16]

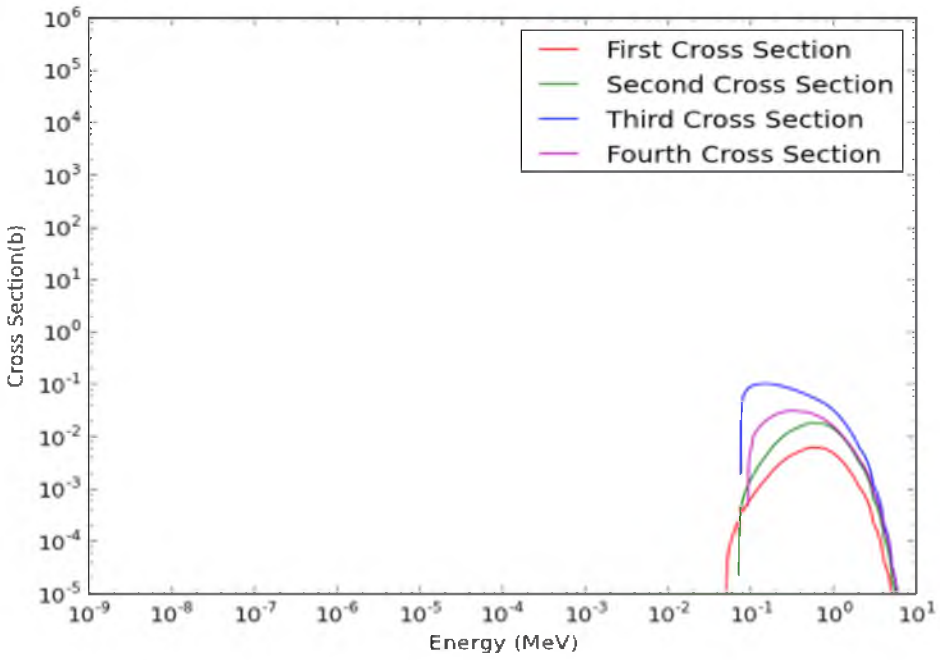

Fig. 2.2: Inelastic scattering cross-sections for Eu-151 [16]

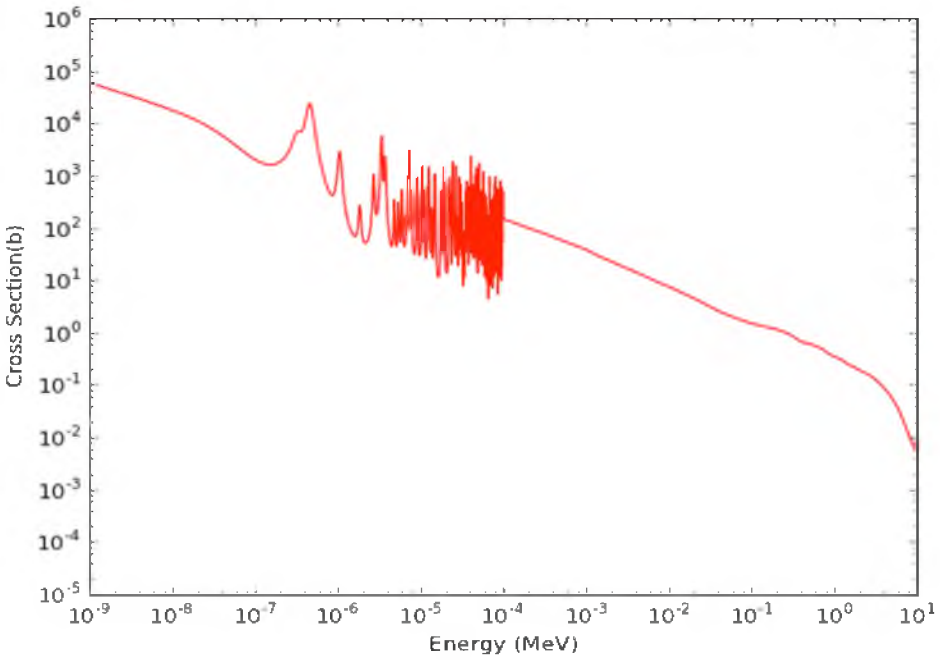

Fig. 2.3: Radiative capture cross-section for Eu-151 [16]

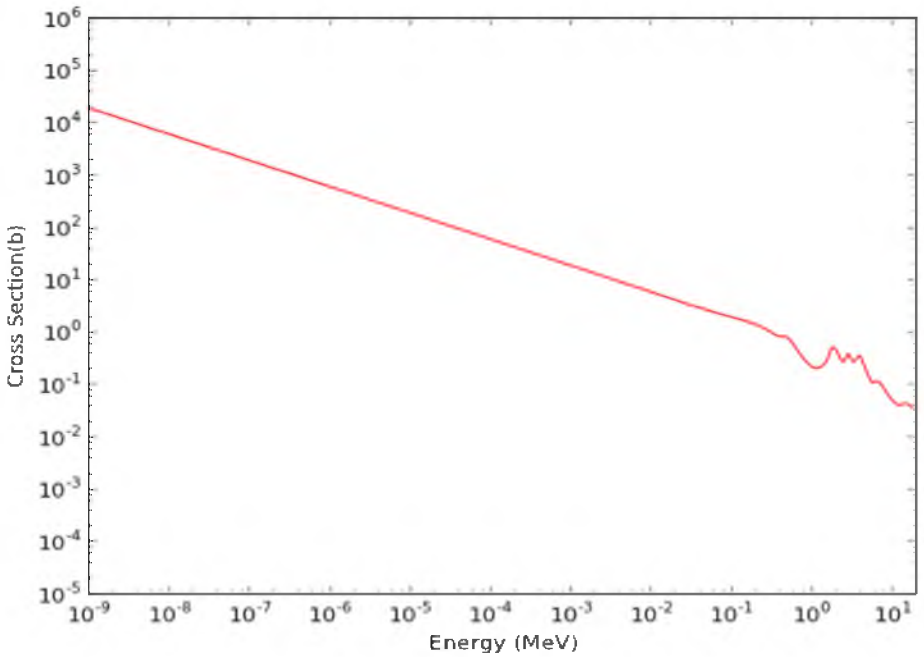

Fig. 2.4: Microscopic cross-section for B-10 + n [16]

CHAPTER 3

INSTRUMENTAL NEUTRON ACTIVATION ANALYSIS

3.1 Introduction

Instrumental Neutron Activation Analysis (INAA) is a nondestructive testing method used to determine elemental composition of a given material. A material is irradiated with neutrons, and the isotopes present are activated by way of neutron capture. The activated isotopes undergo radioactive decay. Different modes of radioactive decay such as beta emission, positron emission, and isometric transition are often accompanied by a gamma-ray. The energy of the emitted gamma-rays is well known and can be uniquely attributed to an individual isotope by measuring their energies on gamma detectors. The process of INAA is illustrated in Fig.3.1 where a neutron interacts with a target nucleus. In order to become stable again, the nucleus emits a prompt gamma-ray and a beta particle along with a delayed gamma-ray.

Using semiconductor detectors such as High Purity Germanium (HPGe) or scintillators such as NaI, the energies of the gamma-rays emitted from the decay are measured. Using software such as Canberra's GENIE 2000, the activity is then calculated, and then concentration of the parent isotope can be calculated.

3.2 The INAA Protocol as Developed in the Nuclear Engineering Program at the University of Utah

Utah Nuclear Engineering Facility (UNEF) houses a 100 kW TRIGA Mark I reactor used for education, training, and research. In order to identify the target nuclides of Co and Eu contained in concrete materials, INAA is used as the primary method. Due to the radioactivity of the samples after neutron irradiation, it is necessary to estimate the potential activity and dose one can receive from any given sample in order to minimize exposure, determine long-term storage needs of the irradiated samples, and increase safety. Being able to identify potential risks associated with irradiated samples also enables researchers to alter experiments as needed to ensure safety. The various irradiation

scenarios are predicted and examined through the INAA Precalculator tool developed by UNEP. Activity (A) is calculated simply as

$$A = N\lambda \tag{3.1}$$

where

$A = activity\,(Bq)$

$N = number\,of\,atoms$

$\lambda = decay\,constant\,(\frac{1}{sec})$

The activity at any given time is calculated as follows:

$$A = A_0 e^{-\lambda t_{decay}} \tag{3.2}$$

where

$t_{decay} = decay\,time\,(sec)$

In order to use the INAA Precalculator, the following information is needed: elements expected to be activated and their respective concentrations in the sample, sample weight, neutron flux (Φ), irradiation time, and decay time. Assuming a mono-energetic neutron flux, the activity of the daughter isotope (A_D) at a given time is calculated as follows:

$$A_D(t) = \phi\Sigma(1 - e^{-\lambda_D t_{irr}}) \tag{3.3}$$

where

$\phi = neutron\,flux\,(\frac{neutrons}{cm^2 \times s})$

$t_{irr} = irradiation\,time\,(sec)$

$\lambda_D = daughter\,decay\,constant\,(\frac{1}{sec})$

Irradiation in the Thermal Irradiator (TI) is assumed to be mono-energetic thermal neutrons, which greatly simplifies the calculations. This, however, is not a real representation of the port environment, which has a broad spectrum of neutron energies. In this case, the range of neutron energies needs to be integrated as follows:

$$R = N \int_0^\infty \phi dE \int_0^\infty \sigma_{absorption} dE \tag{3.4}$$

$$A_D(t) = N_p(1 - e^{-\lambda_D t_{irr}}) \int_0^\infty \phi dE \int_0^\infty \sigma_{absorption} dE \tag{3.5}$$

where

$R = reaction\,rate\,(\frac{neutron\,interactions}{sec})$

$N_P = unit\,number\,of\,parent\,atoms\,(\frac{atoms}{cm^3})$

Equations 3.1 and 3.3 give the daughter activity at t = 0; however, it is necessary to calculate the activity at the time the sample is removed from the UUTR, which can be minutes or hours after sample irradiation. By combining (3.1) and (3.3), the activity of the daughter can be calculated as follows:

$$A_D(t) = \phi\Sigma(1 - e^{-\lambda_D t_{irr}})e^{-\lambda_D t_{decay}} \tag{3.6}$$

The actual activity of the sample is measured using one of three gamma spectroscopy stations utilizing High Purity Germanium (HPGe) detectors. A sample is placed on the detector and measured for a given time, after which GENIE 2000 software from Canberra is used to identify peaks in the gamma spectrum and calculate the activity. Based on the activity the number of parent atoms present in the sample is calculated from the daughter isotope activity using (3.7) and from the granddaughter isotope activity from (3.8).

$$N_P = \frac{A_D(t)}{\sigma_p\phi(1 - e^{-\lambda_D t_{irr}})e^{-\lambda_D t_{decay}}} \tag{3.7}$$

$$N_p(0) = \frac{A_G(t)}{\frac{\sigma_p\phi}{(\lambda_D - \lambda_G}[\lambda_D(1 - e^{-\lambda_G t_{irr}})e^{-\lambda_G t_{decay}} - \lambda_G(1 - e^{-\lambda_D t_{irr}})e^{-\lambda_D t_{decay}}]} \tag{3.8}$$

where

$\sigma_p = cross\ section\ of\ parent\ isotope\ (barns)$

$\lambda_G = Granddaughter\ decay\ constant\ (\frac{1}{sec})$

3.3 Dose Rate Limits

According to NRC Regulations Title 10 Part 20 (10 CFR 20) a radiation area is defined as an area where an individual may receive a dose rate of 1 mrem/hr at a distance of 30 cm from the source[17]. Therefore, samples are not removed from the reactor if they are above a dose rate of 1 mrem/hr at 30 cm. The dose rate is calculated in the INAA Precalculator using the following equation:

$$Dose\ Rate\ (\frac{mrem}{hr}) = \frac{A_D(R)(E)\frac{\mu}{\rho}(1.6 \times 10^{-13}\frac{J}{MeV}) \times (1000\frac{g}{kg}) \times (3600\frac{S}{hr}) \times (10^5\frac{mrem}{\frac{J}{kg}})}{4\pi r^2} \tag{3.9}$$

where

$R = decay\ ratio$

$E = decay\ energy\ for\ gamma\ ray\ (MeV)$

$\frac{\mu}{\rho} = mass\ energy\ absorption\ coefficient\ (\frac{cm^2}{g})$

$r = distance\ from\ source\ (cm)$

3.4 Equipment Used to Measure Dose Rate and Activity

The INAA Precalculator gives a good prediction of what the expected dose will be; however, the actual dose is measured using an ion-chamber detector and is the determining factor in what the dose actually is. An ion chamber is a cylindrical gas-filled detector where one electrode is the cylinder itself and the other electrode is a wire that is insulated from the cylinder. UNEP uses a Ludlum model 9DP-1 with serial number 25006587 and a counting efficiency of 10% shown in Fig. 3.2. As radiation enters the tube, the gas molecules are ionized, which produces positive ions and electrons. An electric potential is applied, and the positive ions are attracted to the anode and the electrons to the cathode, inducing an electric current. The voltage applied to ion chambers creates a large enough difference in current generated by alpha and beta particles that allows them to differentiate between the two particles. The ability to differentiate between the types of radiation allows ion chambers to be used to estimate dose from a sample [18].

HPGe semiconductor detectors are ideal for identifying individual isotopes contained in the sample due to their high resolution compared to NaI or LaBr3 scintillators. An incident gamma-ray interacts with the HPGe crystal and creates primary electrons that have energies greater than thermal energies. This raises electrons' occupied bands from well below the valence band to energy levels above the base of the conduction band, creating holes where the electrons were. The electrons then redistribute themselves until all holes lie at the top of the valence band and the electrons are in the conduction band. In the presence of an electric field, as more electrons are excited and move up, holes move down creating a current. The number of electron hole pairs produced is directly related to the gamma-ray energy absorbed [19]. The current produced is then amplified and sent to a Multichannel Analyzer (MCA) that records the current electric pulses. Each pulse is registered as a count and placed in an energy bin or channel. All the counts are summed for each channel and displayed on a computer connected to the MCA using GENIE 2000 software. As more counts are collected in various channels, these register as peaks in the spectrum and correspond to a specific energy that can be attributed to a specific isotope.

3.5 Sources of Error in INAA Analysis

The sources of errors associated with INAA are dead time, coincidence peaks, and detector calibration. The Analogue-to-Digital Converter (ADC) inside the MCA is only able to handle one pulse at a time. The time at which the ADC is in operation is known as dead-time because no other pulses are able to be processed at this time and hence go undetected. The dead time increases as the activity of a sample increases as more gamma-rays are emitted at any given time. Typically, samples are not counted if their dead time is above 3–5% of the actual time counted.

HPGe detectors have excellent resolution; however, many isotopes emit gamma-rays at very

low energies. Between the energies of 500–501 keV there are around 39 different isotopes that emit gamma-rays [20]. Some isotopes even emit gamma-rays at the exact same energies as others known as coincidence peaks. For example, Po-200 and Xe-141 both emit a gamma-ray at 599.7 keV. This makes it difficult to differentiate between isotopes if there are a number of them present in an analyzed sample. This is overcome by counting samples multiple times. A sample can be counted almost immediately after irradiation and then again several hours or even weeks later. The short-lived isotopes will decay, and long-lived isotopes will remain present and thus more easily detected.

Another source of error comes from detector calibration. Each HPGe detector must be calibrated for energy and efficiency at specific counting geometries. An energy calibration is performed every 1–2 years to set the energy range of the spectrum and correspond known energies peaks with specific channels in the MCA. A multigamma standard is used in this case with known energies. The efficiency calibration is performed after the energy calibration and assigns a gamma emission rate to a specific energy. Due to the random nature of decay, gamma-rays will be emitted isotropically from the sample. Only gamma-rays emitted in the same direction as the detector have a chance of being detected. If only the gammas that are detected are taken into account, the activity of the sample would be grossly underestimated. Samples placed in different geometries will attenuate gamma-rays differently from the calibrated geometry and therefore, efficiency calibrations cannot be used for any other geometry.

At the time of completing this research, calibration standards that are available are old and contain isotopes that emit many different gamma peaks. Too many peaks can interfere with each other and make it difficult for the curve fit algorithm in GENIE 2000 to create an accurate efficiency curve as shown in Fig. 3.3 for counting station number 8 inside UNEF (Utah Nuclear Engineering Facility). There are a total of three gamma spectroscopy stations in UNEF numbered 4, 5, and 8. These each differ in crystal size as well as energy resolution, with station 4 having the broadest energy range.

An isotope whose energy falls on the efficiency curve where there are many data points creates uncertainty as to what the efficiency is. An unknown efficiency will result in erroneous results. Ideally, an efficiency curve should look like the one shown in Fig. 3.4. This efficiency curve was created using a number of radionuclides, each of which emit a gamma-ray at a different energy that does not conflict with the other. Using a gamma source like this creates less data points and allows for a much better best-fit curve, resulting in a more accurate efficiency calibration.

Despite these sources of error associated with HPGe detectors, proper training and experience can keep the errors to a minimum, making INAA an extremely accurate form of elemental analysis.

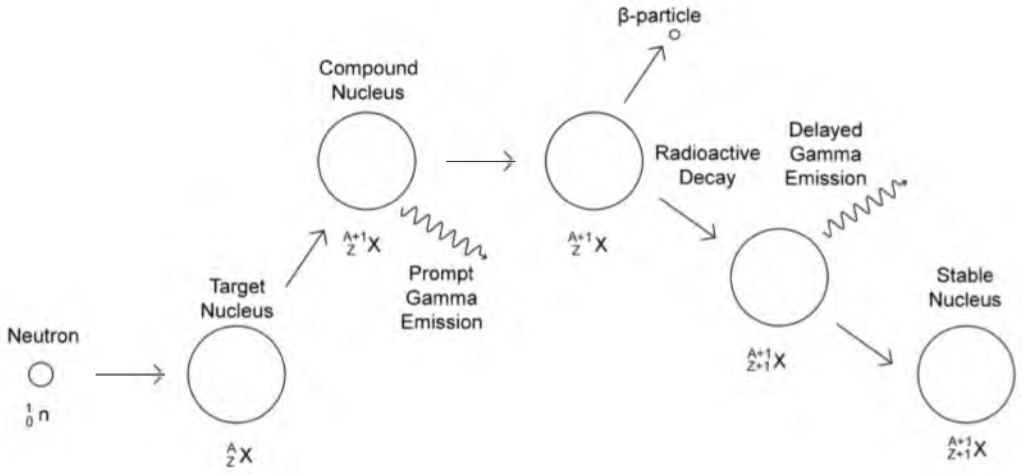

Fig. 3.1: Neutron activation and radioactive decay

Fig. 3.2: Ludlum 9DP-1 ion chamber for dose measurements of irradiated samples

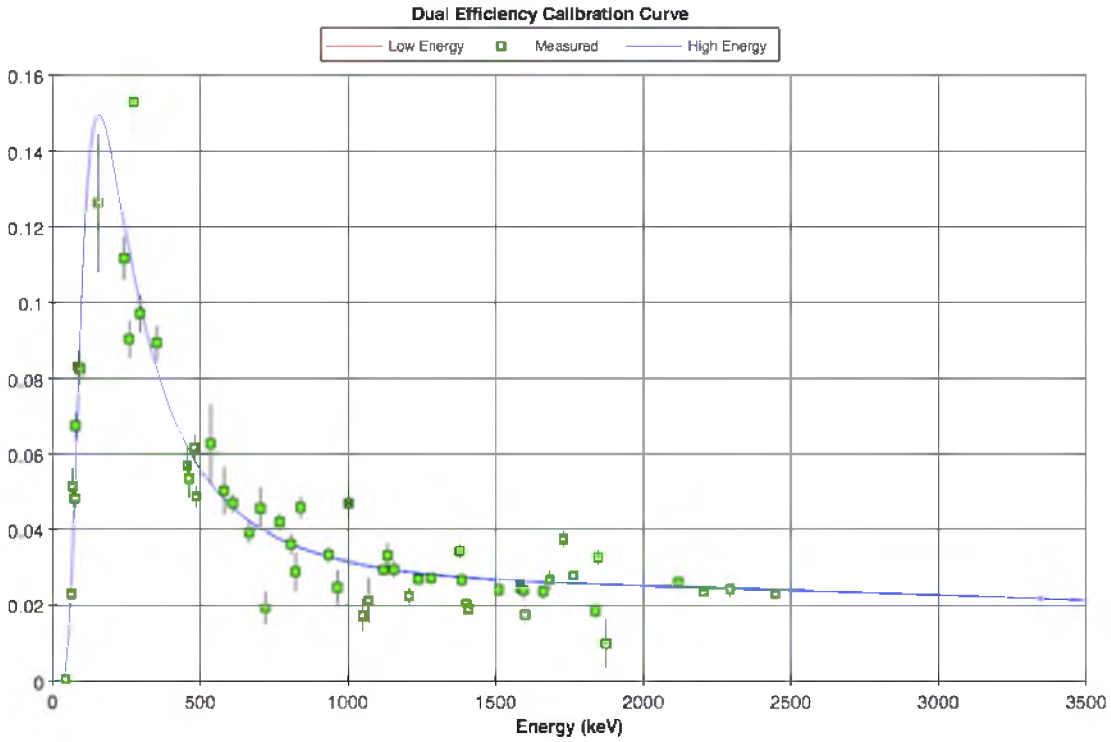

Fig. 3.3: Efficiency curve of Station 8 in UNEF (Utah Nuclear Engineering Facility)

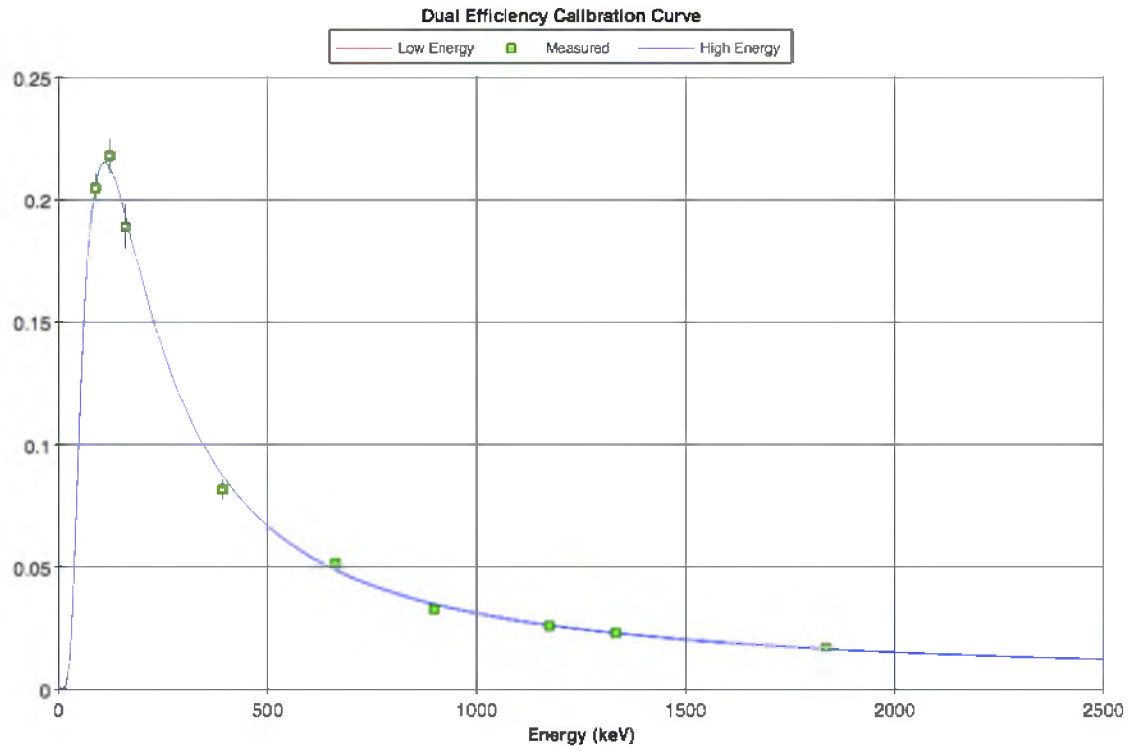

Fig. 3.4: Example efficiency curve of Idaho State University HPGe Detector [21]

CHAPTER 4

INDUCTIVELY COUPLED PLASMA MASS SPECTROMETRY ANALYSIS

4.1 Introduction

Inductively Coupled Plasma Mass Spectrometry (ICP-MS) is a technique used to identify the elemental composition of a sample. ICP-MS is unique due to its extremely low detection limits of many elements in the subparts per trillion (ppt) range [22]. An Agilent 7500ce ICP-MS system is available for use in The University of Utah Department of Geology and Geophysics.

An ICP-MS system consists of three primary region: ionization, ion optics, and mass spectrometer. Samples are delivered to the ionization region in a liquid form. A nebulizer converts the liquid to an aerosol that is delivered to a spray chamber where large droplets are removed from the smaller droplets. The small droplets are transported to the plasma torch that dries, vaporizes, atomizes, and finally ionizes the small droplets. Ion-optics are used to focus the ion beam toward the mass spectrometer. The ion beam passes through a mass analyzer (i.e. quadrupole, magnetic sector, or time-of-flight) where ions are separated by mass before interaction with the spectrometer, where the ions are converted to an electrical signal and counted. The parts of an ICP-MS instrument are outlined in Fig. 4.1 and further explained in this chapter. For simplicity, only a quadrupole mass analyzer will be described in detail.

4.2 Droplet Formation

Droplet formation occurs through usage of a nebulizer and is illustrated in Fig. 4.2. There are two types of nebulizers in use, namely pneumatic and ultrasonic type. The Agilent 7500ce utilizes a pneumatic glass concentric nebulizer. A pneumatic nebulizer consists of a needle, an accelerated propellant gas, and a means to introduce the liquid sample with the gas stream. The gas, usually argon, collides with the liquid, producing an aerosol of various sized droplets. The plasma used to ionize the aerosol is not very efficient at ionizing large droplets. Therefore, it is necessary for the larger droplets to be discarded in the spray chamber to ensure that only small droplets enter the plasma. The spray chamber also functions to smooth out pulses that occur during nebulization [23].

The ionization process of cobalt, closely studied in this research, is explained to better illustrate this process. Each cobalt atom consists of 27 protons, 27 electrons, and 32 neutrons with an atomic mass of 59 Da. As a cobalt atom passes through the plasma, a single electron is stripped creating a positively charged cobalt ion. This ion still consists of 27 protons and 32 neutrons and a mass of 59 Da but now has 26 electrons, creating the positive charge. Some elements, like europium, have two naturally occurring isotopes (i.e., Eu-151 and Eu-153). As Eu atoms are ionized, some will have a weight of 151 Da and others 153 Da. These unique masses are eventually measured by the mass spectrometer and counted to quantify the elemental composition of the sample.

4.3 Interface Region

The plasma is kept at atmospheric pressure or 101.3 kPa, while the mass spectrometer is kept at a significantly lower pressure of 1x10-4 Pa [24]. The purpose of the interface region is to efficiently transport the ions created in the plasma to the mass spectrometer. The ions pass through a series of cones with small orifices that descend in size. The interface region generally consists of 2–3 cones with the last one having a diameter of 0.4–0.5 mm [24].

4.4 Ion-Focusing System

As the ions travel from the atmospheric pressure plasma to the mass spectrometer under vacuum, they will have a tendency to disperse and scatter. The ion-focusing system is made up of a series of metallic plates that have a voltage applied to them. The plates act to steer the ions that have a tendency to disperse while traveling out of the interface region towards the mass analyzer and in a narrow, focused beam.

Overall, the ion focusing system is inefficient at getting the ions to the mass spectrometer. Only about 1 in 1 million ions will actually reach the detector for analysis [25].

4.5 Mass Analyzer

The mass analyzer is positioned between the ion optics system and the detector. A quadrupole mass analyzer is kept under vacuum at 1x10-4 Pa. The 7500ce instrument utilizes a quadrupole mass analyzer that consists of four cylindrical metal rods made of either steel or molybdenum. The quadrupole is used to separate the ions by mass and filter a selected mass range to pass to the detector. This is accomplished by having a DC current and AC current on opposite rods. As the AC/DC ratio changes, particles of different masses will either become unstable and be ejected or be allowed to pass through uninhibited for detection. The ions that pass through will then interact with the ion detector and be converted to an electrical pulse and counted. Typically, a mass analyzer can process 25 elements in 2–3 minutes [26].

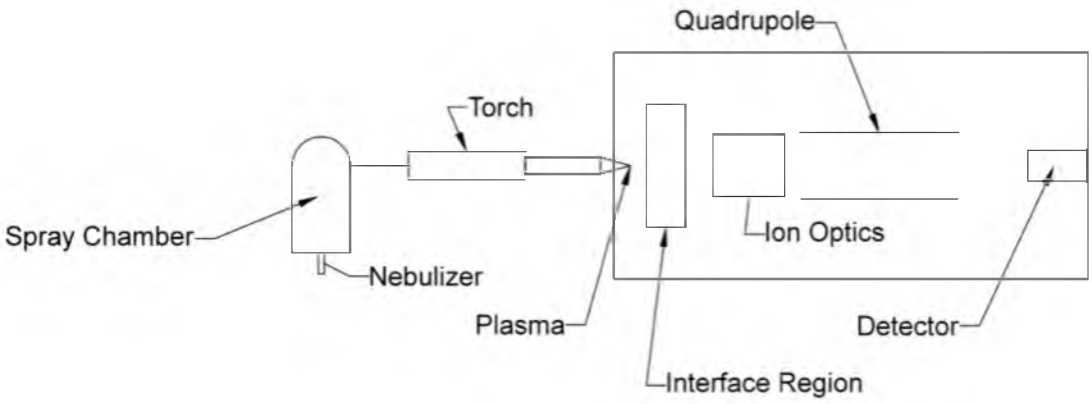

Fig. 4.1: Overview of ICP mass spectrometer

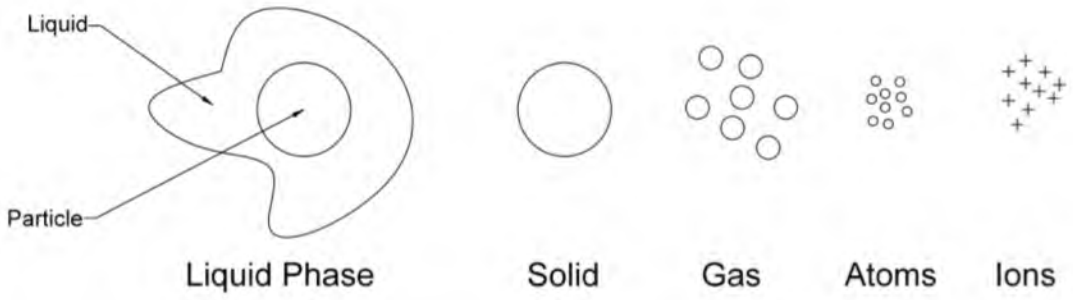

Fig. 4.2: Phase changes of sample while passing through plasma

CHAPTER 5

ESTABLISHING THE COMPOSITION OF RADIATION RESISTANT CONCRETE: NAVA ALIGA CHARACTERISTICS

5.1 Concrete Chemistry

NAVA ALIGA (ancient word for a "new stone") is a new concrete mix concept that has resulted from research conducted at the Utah Nuclear Engineering Program. The research on this mixture starting in 2011 with graduate student Jason Rapich, who established the protocol based on INAA to analyze concrete aggregates and cement for Co and Eu content. The mixture is intended to be a long-term durable concrete with the same properties as concrete used in currently operating nuclear power plants, but allows for it to be recycled as regular concrete rather than radioactive waste. NAVA ALIGA will never be activated above the IAEA clearance level after long-term exposure to neutron radiation when used as a material for reactor dome, spent fuel pool, or radioactive waste canisters.

Concrete is primarily composed of four main ingredients: water, cement, fine aggregate, and coarse aggregate. The cement and water primarily contribute to the compressive strength of concrete. Cement acts as a binder when mixed with water and allowed to go through the hydration process. During hydration, which can occur over several years, concrete will densify as the cement continues to generate products and fill in remaining void spaces. While water is a necessity for hydration to occur, the overall strength of concrete is directly related to the water to cementitious (w/cm) ratio. A minimum w/c ratio of 0.40 to 0.50 is required for nuclear safety-related concrete structures [27]. Liquid chemical admixtures such as a "water reducer" like lignosulfonate, polycarboxylate ether, and polymelamine sulfonate can be added to increase workability, or can be used to decrease the w/c ratio and increase compressive strength.

Cement is primarily composed of the oxides shown in Table 5.1. Additional trace elements are also found, such as cobalt and europium, that do not contribute to the strength of concrete. These trace elements are also found in the coarse and fine aggregates and are what have been found to

contribute to the long-term radioactivity of activated concrete.

Other admixtures can be "pozzolanic" materials, such as fly ash, silica fume, or blast furnace slag, and can be added to concrete in order to increase characteristics such as strength and density or lower the permeability. Pozzolanic materials have a high SiO2 content and when used in conjunction with water and Calcium Hydroxide (CH) formed from cement, Calcium-Silica-Hydroxide (C-S-H) gel is formed, which is primarily responsible for concrete strength. Each material component in concrete (aggregates, cement, admixtures) must be investigated and elemental composition determined in order to verify that they do not contain the Co and Eu isotopes identified as contributing to the long-term radioactivity of activated concrete. These ingredients are investigated using two methods: INAA and ICPMS.

Collaboration with a local concrete supplier provided samples of a variety of components used in concrete for irradiation that were currently in use for various construction projects around Utah's Wasatch Front. Many of these samples had similar characteristics as components Japanese scientists identified as being suitable for low-activation concrete, namely, white cement and aggregates with minimal processing [6]. The samples types and number of each that underwent testing are outlined in Table 5.2, as well as the testing outlined in Table 5.3.

These components were tested using INAA by UNEP in 2011 in order to identify samples with low target nuclide concentrations [28]. Of the 39 samples tested in total (see Table 5.2 below for breakdown of what types of materials), only one white Portland cement, one silica sand, and ones coarse aggregate were identified as having low-activation characteristics, shown in Fig. 5.1. These three samples were further studied using INAA (for exact Eu and Co amounts) as well as ICPMS to verify results.

INAA was chosen as the primary method for quantifying the trace elements Eu and Co. INAA has low detection limits as well as the ability to detect many of the heavier elements and their isotopes on the periodic table. Samples were stored in 5-gallon buckets (one bucket of coarse aggregates, one bucket for the fine aggregates, and one bucket of the cement). Five representative smaller samples were manually pulled from each bucket to be irradiated. The irradiation port in the reactor is small, so all samples needed to be in a powder form. The coarse aggregate was therefore crushed and blended into five samples that fit the size constraints of the TI. The same crushing and blending procedure was applied to the fine aggregate. Each analysis sample mass was on average 1 g, and each sample was irradiated in the UUTR for 101 minutes at a thermal power of 90 kW, which provides a neutron flux of 3.76x1011.

Eu-152 emits 148 different gamma-rays of varying intensities ranging from 5.64 keV to 1769.09 keV [20]. Of these 148 only 18 are emitted with an intensity greater than 0.01, as indicated in Table

5.4. The most prominent of these are only a few hundred keV in energy and fall in the range with the most background interference that can drown out the Eu-152 signal and make it difficult to detect. Co-60 emits six different gamma-rays, shown in Table 5.4, with two prominent energies accounting for over 99.99% of all emissions[29].

After irradiation, all samples were allowed to decay for a period of 5 months so that short-lived isotopes had a chance to decay away and therefore eliminate any spectrum interference created from radionuclides emitting gamma-rays at similar energies. Each analyzed sample was counted on an HPGe detector for a period of 24 hours in point source geometry. Following this, all of the irradiated samples that were counted at UNEF were sent to Environmental Monitoring Laboratory (EML) at Idaho State University (ISU) to be counted on their HPGe detectors for additional verification of the quantified radionuclides emitted.

Using GENIE 2000 software developed by Canberra, the samples were analyzed for the presence of Co-60 and Eu-152. The activities of each of the detected isotopes are output, and the elemental concentration is calculated using (3.6) or (3.7). Both of these equations require the half-life of the isotope and therefore take into account the differences in decay time between counting and UNEF and EML.

5.2 Inductively Coupled Plasma Mass Spectrometry

Additional crushed samples of the three candidate materials were also analyzed using an Agilent 7500ce ICP-MS at the University of Utah Department of Geology and Geophysics as a third testing method to verify Co and Eu.

Each of the three samples were then washed in ultrapure Milli-Q water in order to remove and surface contaminants. Samples were then digested in a single bath of ultrapure HCl, HNO3, and HF acids. All three samples were anticipated to contain high amounts of Ca and Si and were therefore diluted so as to decrease the spectral interference during mass spectrometry. This is due to the low detection limits of ICP-MS (parts per trillion) that can become overwhelmed with a high concentration of material. The diluted samples were then loaded into a Cetac AS 520 auto sampler that delivered the samples to an Agilent 7500ce mass spectrometer. The results were presented to UNEP following analysis.

5.3 Elemental Composition as Determined From INAA and ICP-MS

Table 5.5 shows a large discrepancy between INAA values, compared to the values obtained from ISU or ICPMS. While the results from ISU do not match exactly with ICPMS values, they do fall within the same order of magnitude and both methods could successfully detect both Co and Eu.

Fig. 5.2 shows the finding that all samples fell below IAEA clearance levels except for europium content of fine aggregate as measured by Idaho State University. Coarse and fine aggregate as well as the components of cement are heterogeneous materials. The composition of the samples may be different despite selecting sample from the same aliquot as a 1 gram sample may not be representative of the entire content of materials. This could explain the small differences between the samples measured at EML and those using ICP-MS.

Each of the UNEF HPGe detectors was calibrated using a multigamma source calibrated in 2005 with the isotopes listed in Table 5.6. Using the multigamma source, calibration curves were established for both energy as well as efficiency. As shown in the efficiency curve of Fig. 5.3, it becomes very difficult to establish a best-fit curve when there are many scattered data points. The efficiency at any given energy could vary by as much as 2–4%. When trying to detect trace elements in the ppm and ppb range, 2–4% carries much more weight than it would for more abundant elements.

Reducing the number of data points used to create the best-fit curve changes the efficiency used to calculate the activity of measured samples. This is demonstrated in Fig. 5.4, which shows how efficiency deviates from the GENIE 2000 curve as data is reduced to 29 and then to 10 data points. The area of greatest deviation occurs in the energy ranges of 500–1500 keV. This is important as the decay energies of both Eu-152 and Co-60 occur in this range.

Ideally, an efficiency curve should look like the one shown in Fig. 5.5. This efficiency curve was obtained using a number of radionuclides, each of which emit a gamma-ray at a different energy that does not conflict with the other. Using a gamma source like this creates less data points and allows for a much better best-fit curve, resulting in a more accurate efficiency calibration.

Cobalt emits two gamma-rays at energies of 1173 keV and 1332 keV, while europium's highest intensity gamma-rays occur at the energies of 344 keV and 778.9 keV, respectively. Each of these elements' gamma-rays decay in the energy ranges with the greatest discrepancy of UNEP's efficiency curve. It is therefore likely that one reason for the discrepancies in data between UNEP and ISU can be attributable to a poor efficiency curve.

The activity of the multigamma source used for calibration of UNEP HPGe detectors was last verified in 2005 and could itself be a source of error in the gamma spectrometry analysis. The most recent point source geometry calibrations were performed in 2012, 7 years after the last calibration of the multigamma source. Four of the elements listed in Table 5.6 have half-lives less than 1 year and would have undergone over 7 half-life periods since the last calibration. This results in a very weak gamma source that does not provide strong, distinct energy peaks for calibration.

Another important aspect of any calibration is the geometry used during calibration. A densely packed sample of a few hundred grams is going to attenuate its own gamma-rays differently than a loosely packed soil sample of a few milligrams. At UNEP and ISU, a button source was used for

calibration of the detectors. The button source is assumed to be roughly equivalent to a point source. Each of the concrete material samples irradiated was less than 1 g in size and roughly the same size as a button source. The assumption was therefore made that they could also be treated as a point source. This assumption has been made in past INAA analysis and has proven to be an accurate approximation in previous analysis.

Concrete, being primarily composed of rock, sand, cement, and water, is a very heterogeneous material. Likewise, rock, sand, and cement are also heterogeneous in nature. For these reasons standards have been established in order to accurately classify aggregates used in concrete. Due to restrictions in sample size based on the size of the irradiation ports of the UUTR, it is impossible to measure that large a quantity of material at any given time. Even though samples were taken from the same sampling of concrete materials for both ICP-MS and INAA, the heterogeneous nature of these materials is the likely cause of the difference in data between ICP-MS and the gamma spectrometry results from ISU.

5.4 Activity Calculation Following 40-year Neutron Irradiation

Using the NAA Precalculator, the concentrations determined using both NAA and ICP-MS were input in order to determine the activities following a 40-year exposure to neutron irradiation. Japanese research indicates that the inner wall of a nuclear power plant containment dome is exposed to a neutron flux of 2.5×10^5 neutrons/($cm^2 \times s$) during the time of operation [5]. Nuclear power plants are typically issued licensed to operate for 40-years. Figs. 5.6, 5.7, and 5.8 show calculated activities for target radionuclides immediately following 40-years of irradiation and up to 100-years of decay time.

In Figs. 5.7 and 5.8, Eu-152 activities determined from gamma spectroscopy are 1% above the IAEA clearance levels following 40-year neutron irradiation. This is likewise observed with Co-60 determined from ICP-MS in Fig. 5.8, where the activity of Co-60 is 40% above clearance level. This is not a concern because a nuclear reactor is allowed to cool for a period of 6 years before decommissioning begins [4]. In every instance where activities are above IAEA clearance levels, they have adequate time to decay below clearance levels at the time decommissioning begins.

5.5 Natural Radioactivity Comparison Between NAVA ALIGA and Concrete From Zion Nuclear Power Plant

Zion Nuclear Power Station ceased power generation in February 1998 and operated for 24 years. Concrete from the outer portion of the containment dome was sent to UNEP for analysis and

comparison to NAVA ALIGA. Neutron radiation is limited to the first 100 cm of the containment dome, so the outer portion should so no neutron activation products. Zion concrete was counted for a period of 24 hours using the HPGe detector at station 8 in UNEF. NAVA ALIGA that had not been activated was also counted for the same period of time at station 8. The results of this analysis are summarized in Table 5.7.

All the radionuclides listed are naturally occurring and are a result of the decay of uranium and thorium isotopes. This proves that Zion concrete that existed outside of the first 100 cm of the inner wall was not activated by neutron radiation.

Table 5.1: Primary oxides that are present in and vital to the manufacturing of cement

Oxide	Approximate %
CaO	64%
SiO_2	21%
Al_2O_3	5%
Fe_2O_3	3%
SO_3	3%
Na_2O	<0.60%
K_2O	<0.60%

Table 5.2: Number and type of concrete components first tested using INAA by UNEF

Sample Type	Number of Samples Tested of Each Type
Portland Cement	10
Fly Ash	9
Silica Fume	2
Slag	2
Silica Sand	10
Coarse Aggregate	6

Table 5.3: Timeline of testing performed on candidate samples

Test	Date	Purpose
INAA	2011	Initial identification of samples with low target radionuclide concentrations
INAA	February–July 2013	Further evaluation of samples identified from initial testing
ICP-MS	March 2013	Additional verification of the level of trace elements
Gamma spectrometry at EML	September 2013	Additional verification of the level of trace elements

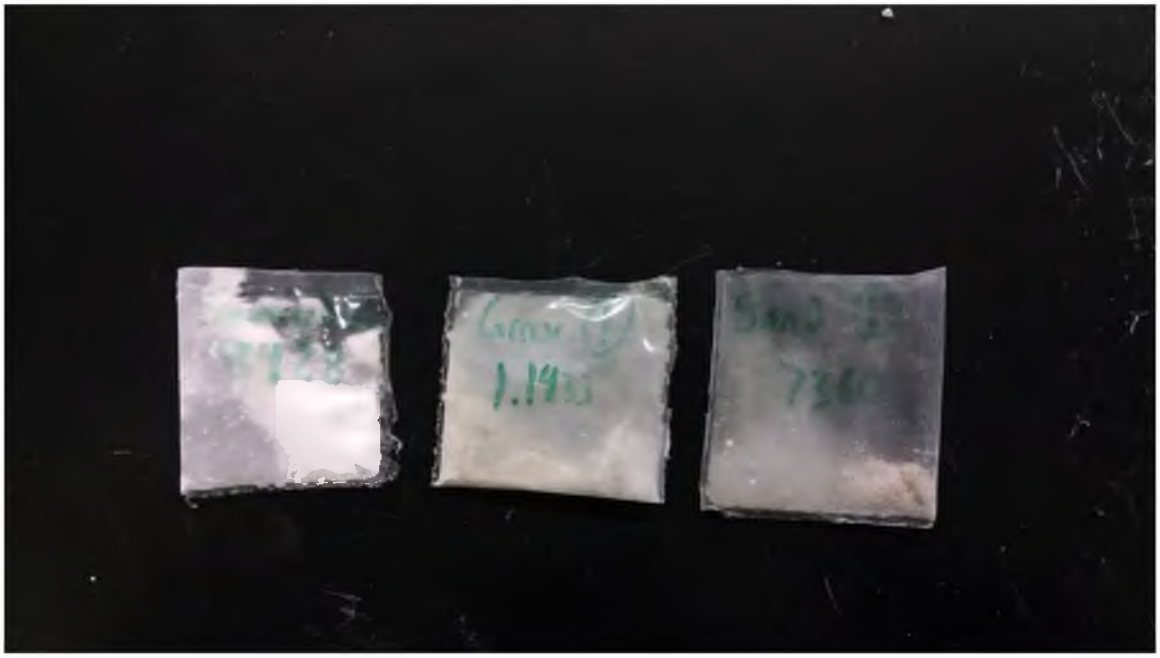

Fig. 5.1: The only materials identified with trace element content below IAEA clearance levels: one cement, one coarse aggregate, and one fine aggregate sample, as prepared for INAA testing

Table 5.4: Gamma-rays emitted by Eu-152 and Co-60 with an intensity greater than 0.01

Nuclide			
Eu-152		Co-60	
Energy	Intensity	Energy	Intensity
40.1181	0.3800	1332.5010	0.9999
121.7825	0.2840	1173.2370	0.9997
344.2810	0.2660		
39.5224	0.2100		
1408.0110	0.2087		
5.6400	0.1510		
45.4000	0.1480		
964.1310	0.1434		
1112.1160	0.1355		
778.9030	0.1296		
1085.9140	0.0992		
244.6989	0.0749		
867.3880	0.0415		
443.9760	0.0278		
411.1150	0.0223		
1089.7000	0.0171		
1299.1240	0.0163		
1212.9500	0.0140		

Table 5.5: Co and Eu content comparison between three different analyses of aggregates and cement

Coarse Aggregate						
	NAA		Mass Spec		ISU Gamma Spec	
Element	Concentration (ppm)	Uncertainty (2σ)	Concentration (ppm)	Uncertainty (2σ)	Concentration (ppm)	Uncertainty (2σ)
Eu	Undetected	-	0.037	2.469E-03	0.049	0.004
Co	Undetected	-	1.050	0.057	0.464	0.022
Sand						
	NAA		Mass Spec		ISU Gamma Spec	
	Concentration (ppm)	Uncertainty (2σ)	Concentration (ppm)	Uncertainty (2σ)	Concentration (ppm)	Uncertainty (2σ)
Eu	0.057	0.012	0.014	9.376E-04	0.105	0.010
Co	Undetected	-	1.850	0.048	0.755	0.079
Cement						
	NAA		Mass Spec		ISU Gamma Spec	
	Concentration (ppm)	Uncertainty (2σ)	Concentration (ppm)	Uncertainty (2σ)	Concentration (ppm)	Uncertainty (2σ)
Eu	0.003	0.010	0.084	3.531E-03	0.034	0.003
Co	0.029	0.094	1.440	0.082	0.256	0.155

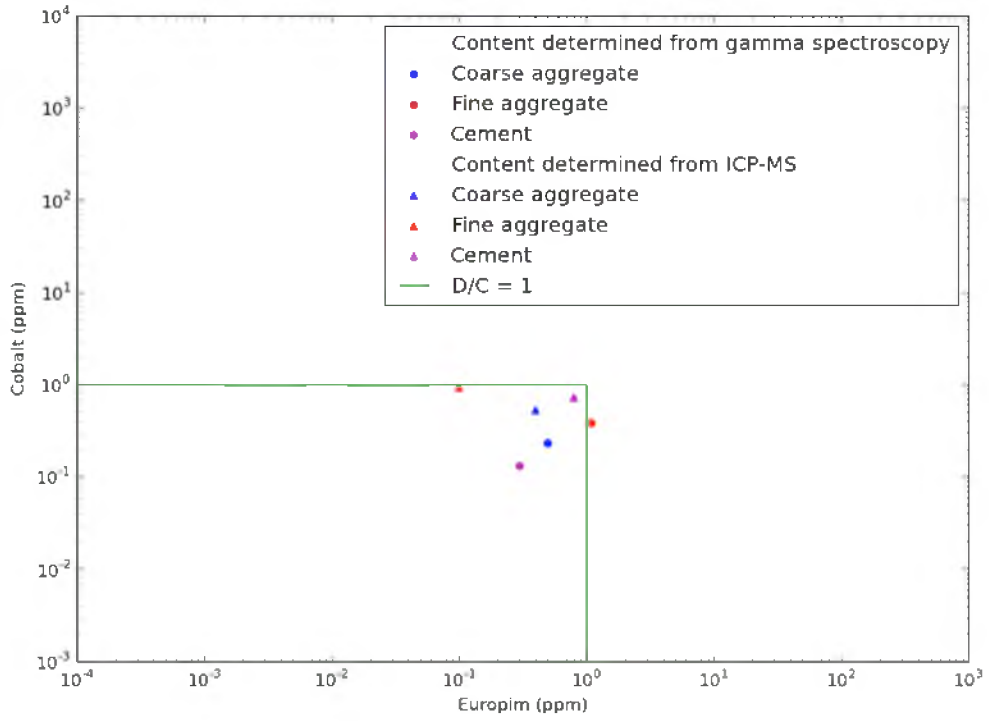

Fig. 5.2: A comparison of cobalt and europium in relation to their clearance levels (D/C = 1)

Table 5.6: Isotopes present in multigamma calibration source

Isotope	Half-Life (years)
Co-57	0.745
Cs-137	30.070
Eu-155	4.761
Mn-54	0.856
Sn-113	0.315
Zn-65	0.669

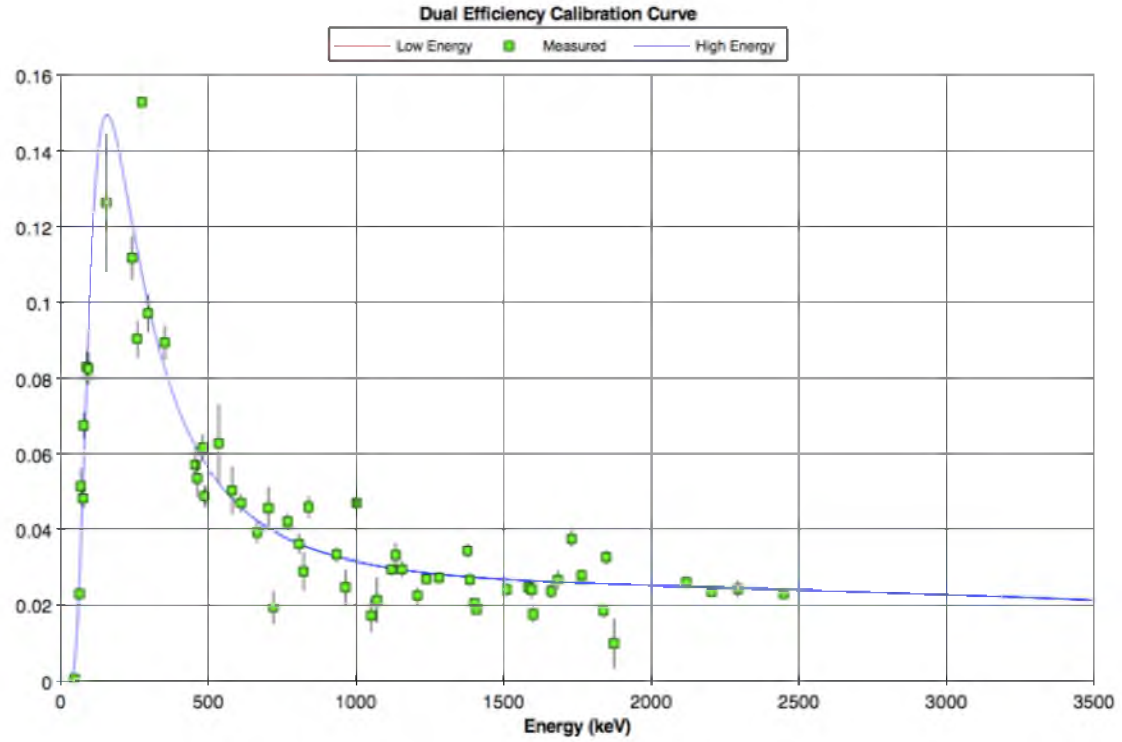

Fig. 5.3: Efficiency curve of Utah Nuclear Engineering Facility radiation counting station 8 HPGe detector

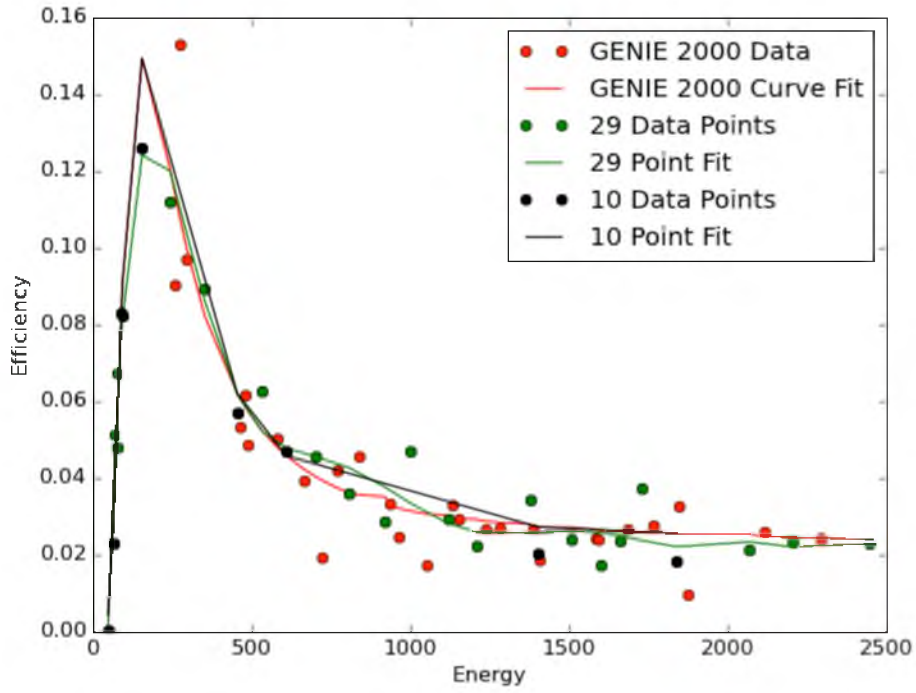

Fig. 5.4: Best fit efficiency curves of radiation counting station 8 in Utah Nuclear Engineering Facility with reduced number of data points

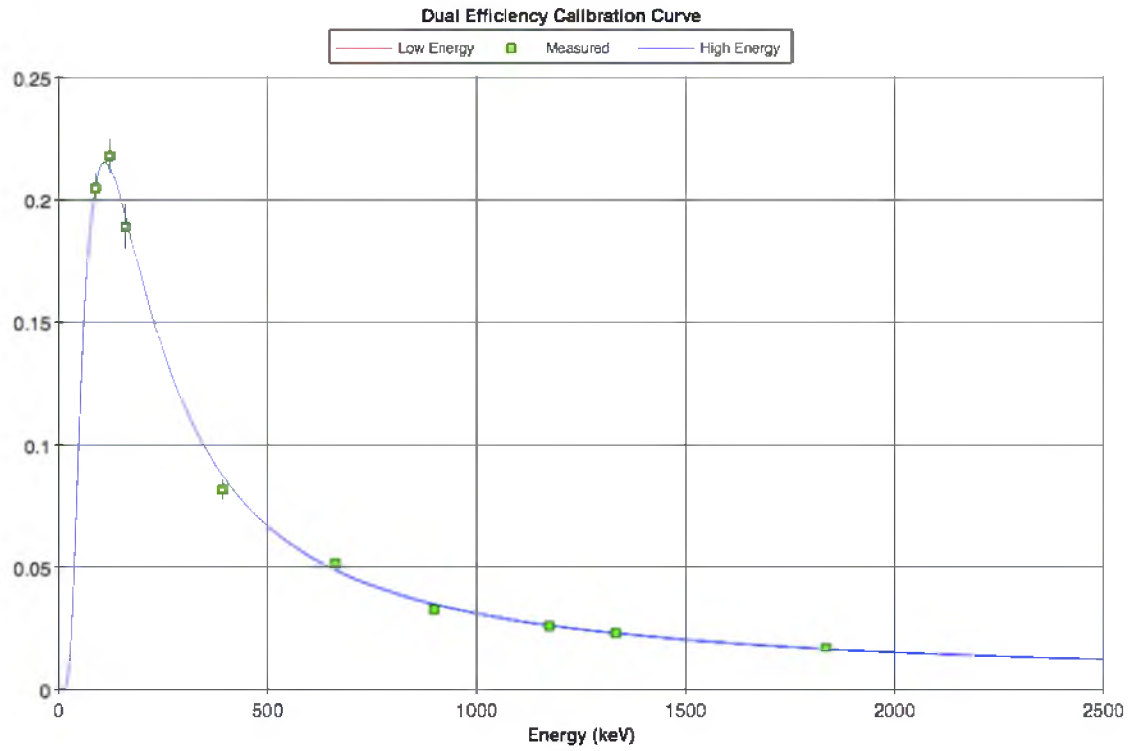

Fig. 5.5: Example efficiency curve from Idaho State University HPGe detector [21]

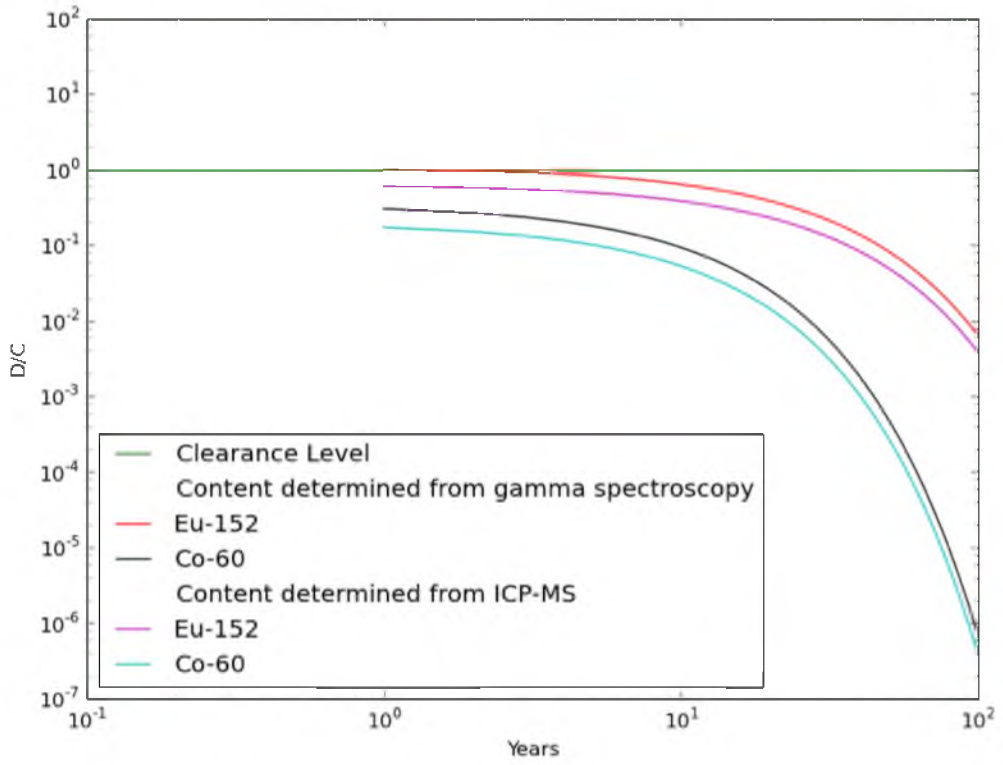

Fig. 5.6: Activities of isotopes identified in cement after calculated 40-year neutron irradiation and a decay time over 100-years

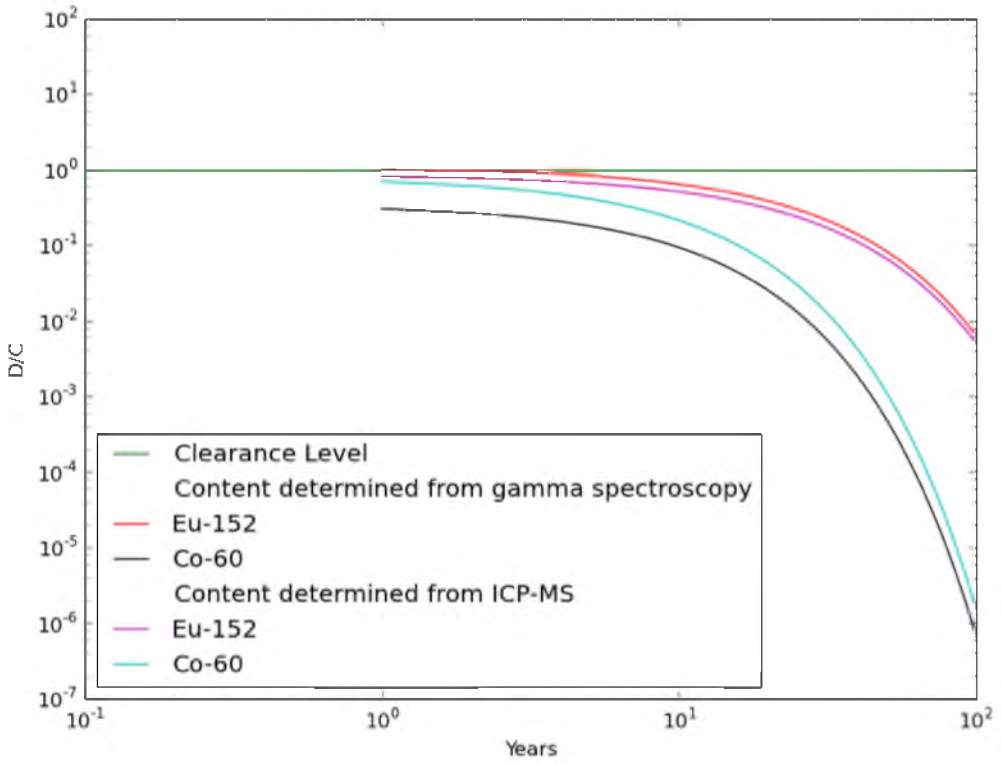

Fig. 5.7: Activities of isotopes identified in fine aggregate after calculated 40-year neutron irradiation and a decay time over 100-years

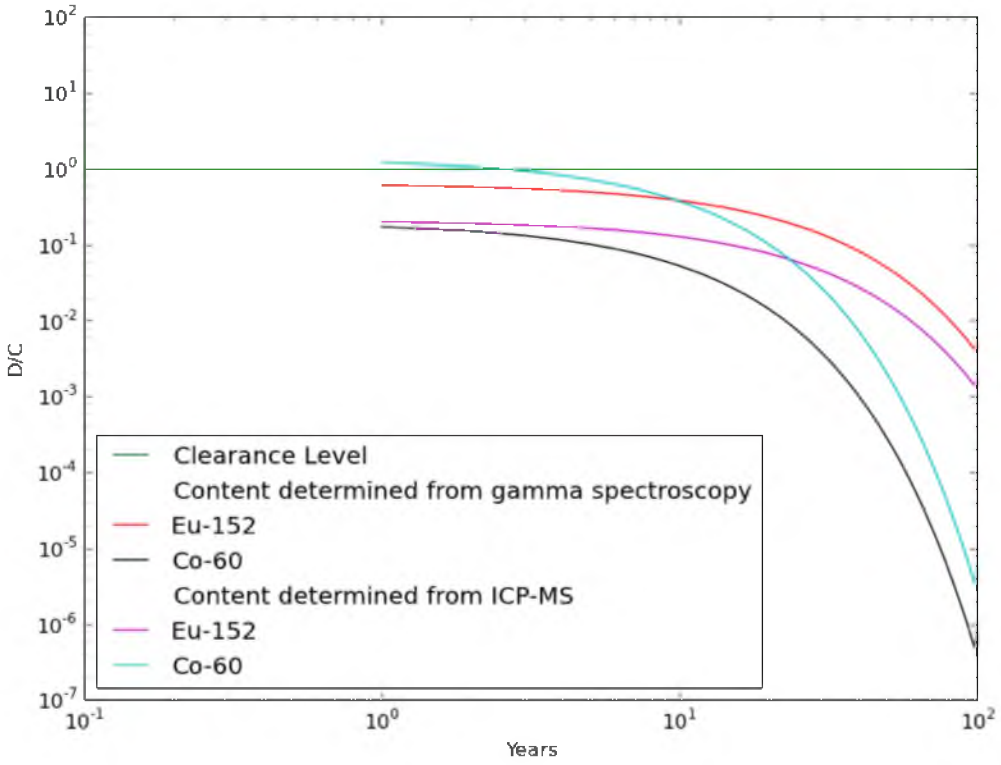

Fig. 5.8: Activities of isotopes identified in coarse aggregate after calculated 40-year neutron irradiation and a decay time over 100-years

Table 5.7: Natural radioactivity occurring in both Zion concrete and NAVA ALIGA

	Zion Concrete	NAVA ALIGA
Nuclide Name	Concentration (ppm)	Concentration (ppm)
Pb-212	4.553E-18	4.637E-18
Pb-214	7.079E-18	5.672E-18
Ra-224	1.721E-17	3.128E-18
Th-231	6.354E-18	4.494E-18
Th-234	1.016E-17	6.952E-18
U-234	1.754E-16	8.758E-17
U-235	1.530E-18	1.175E-18

CHAPTER 6

PHYSICAL PROPERTIES OF NAVA ALIGA

6.1 Original Mix Design of NAVA ALIGA

Following identification of NAVA ALIGA components presented in Chapter 5, a mix design was created to subject the concrete to an array of tests to determine its performance. A local concrete supplier provided the candidate materials and also provided the original mix design that consisted of the proportions listed in Table 6.1 for 1 ft^3 of NAVA ALIGA. Typically, admixtures such as water reducer are added to a concrete mixture in order to decrease viscosity with a minimal amount of water. Since no analysis had been performed on such water-reducing admixtures, instead an additional 5 lbs of water and 2 lbs of cement were added to decrease the viscosity and allow thorough mixing of ingredients with the final mix design in Table 6.1.

The w/c of the final adjusted mixture was 0.56, which is relatively high for structural applications. The actual proportions of concrete components have no bearing on the low-activation characteristics of NAVA ALIGA.

Standard 4x8" concrete cylinders were cast in accordance with ASTM C31 from the candidate materials for both compressive strength and permeability testing. These cylinders were stripped from their plastic molds after 24 hrs and cured by placing the concrete underwater for 27 additional days. A moist-curing environment is a common practice in order to maximize the hydration process.

6.2 Compressive Strength Testing

A total of three cylinders were tested in the University of Utah Structures Laboratory for compressive strength using ASTM C873. Cylinders were placed in a INSTRON universal testing machine, shown in Fig. 6.1, where they were compressed at 67 290 lbs/min until failure (Fig. 6.2). The ultimate strength is observed to be greater than 6 000 psi in every instance.

The resultant ultimate compressive strength was on average 44.2 MPa or 6 411 psi, with the individual results outlined in Table 6.2. For comparison, standard concrete used for general construction purposes such as sidewalks has an average compressive strength of 27.6 MPa or 4 000 psi. Concrete

used in nuclear safety-related structures is required to have a compressive strength of at least 2,500 psi [28].

6.3 Permeability

A growing issue in aging concrete structures is the durability and longevity of concrete. The concrete in a biological shield of a power reactor or in spent fuel pools need to retain water for several decades in order to remain functional. Permeability tests are needed to ensure new concrete mixes are not going to leach the material being contained into the environment.

The PROOVE'it Rapid Chloride Permeability Test (RCPT) is one of the quickest and most effective permeability tests for concrete. It is used to evaluate the diffusivity of chloride ions through concrete following guidelines set by ASTM C1202. The more permeable concrete is, the more chloride ions will migrate through concrete. The migrating chloride ions generate a current that is measured. A higher current indicates higher permeability [30].

This test requires cured concrete in a cylindrical form cut to 100 mm diameter and 50 mm in height. The specimen preparation includes an epoxy coating and complete water saturation. The epoxy is applied to the cylindrical wall to prevent chloride ions from leaching out the sides of the cylinders and only allows the ions to migrate in one global direction. After the epoxy has cured the specimens are placed in a vacuum chamber set to -600 psi. After 1 hour, water is let into the vacuum chamber to submerge the concrete for 3 hours and saturate the specimens. A saturated specimen allows for a current to pass through the specimen.

The saturated concrete is placed into watertight cells that have separate chambers on either side for NaCl and NaOH solutions, of 3% by weight and 0.3 N in concentration, respectively. A 60 V DC potential is applied across a specimen with the negative potential on the NaCl side and the positive terminal on the NaOH side. The negatively charged chloride ions move towards the NaOH side. The current is measured in Coulombs and recorded for a 6-hour period. The diffusivity of the concrete is then classified based on its Coulomb value from Table 6.3, where a higher Coulomb value is indicative of increased diffusivity and permeability.

Four NAVA ALIGA specimens were tested using the RCPT method created from a 4x8" cylinder. The tests are shown in Table 6.4. Due to the increase of diffusivity with the increase in temperature, PROOVE'it provides the adjusted charge passed to account for the increased diffusivity. The test showed NAVA ALIGA to have a high diffusion potential, which is not ideal for usage in the nuclear industry. The w/c was calculated to be 0.56, which is typical of highly permeable concrete.

Table 6.1: Established mix design of NAVA ALIGA

	Original Mix Design	Adjusted Mix Design
Material	Weight (lbs.)	Weight (lbs.)
Cement	26.1	28.1
Water	10.8	15.8
Fine Aggregate	55.1	55.1
Coarse Aggregate	55.1	55.1

Fig. 6.1: NAVA ALIGA cylinder in compressive strength testing apparatus

Fig. 6.2: NAVA ALIGA cylinder post compressive strength testing

Table 6.2: Maximum compressive strength of NAVA ALIGA cylinders

Cylinder	28 day Compressive Strength MPa (psi)
1	44.8 (6,499)
2	45.1 (6,543)
3	42.8 (6,219)

Table 6.3: Permeability classes based on Coulomb values [30]

Coulombs	Permeability Class	Typical w/c values
> 4,000	High	> 0.5
4,000 - 2,000	Moderate	0.4 - 0.5
2,000 - 1,0000	Low	< 0.4
1,000 - 100	Very Low	Latex modified concrete
< 100	Negligible	Polymer concrete

Table 6.4: Measured RCPT values for NAVA ALIGA

Sample No.	1	2	3	4
Charge Passed (Coulombs)	6946	6555	6745	4909
Adjusted Charge Passed (Coulombs)	6269	5916	6087	4430
Permeability Class	High	High	High	High

CHAPTER 7

USE OF DEPLETED URANIUM WITH NAVA ALIGA FOR ENHANCED SHIELDING CAPABILITIES

7.1 Properties of Depleted Uranium

Uranium is present throughout soil at a concentration of 3 ppm and occurs in the form of three different isotopes, U-234, U-235, and U-238, with approximately 99.27% of natural uranium being U-238 [16, 31]. Of the three naturally occurring uranium isotopes only U-235 is considered fissile. As a result, the concentration of U-235 must be increased for usage in commercial power plants, naval reactors, and nuclear weapons in what is known as the enrichment process. The enrichment process creates an abundance of what is known as Depleted Uranium (DU). While DU is primarily composed of U-238, it does contain about one-third the naturally occurring concentration of U-235 as well as other trace elements form the enrichment process [31]. DU is primarily an alpha emitter and is not considered a risk for external exposure to humans due to low penetration of alpha particles. Uranium is 65% more dense than lead, which is frequently used for shielding against radiation [31]. DU therefore has great potential to be used as a more effective biological shield in hospital and waste disposal sites.

7.2 MCNP6-Based Model of Depleted Uranium With NAVA ALIGA

Due to the toxicity of depleted uranium, it was necessary to first model its addition with NAVA ALIGA using MCNP6. MCNP, which stands for Monte Carlo N-Particle is a general-purpose radiation transport code [32]. In order to understand how much more effective DU is when combined with concrete as a biological shield, a variety of different simulations were developed. MCNP6 was used to test the shielding properties of concrete (NAVA ALIGA), depleted uranium, lead, and a combination of concrete and DU.

An MCNP6 model consisting of a 5-m block made of concrete, lead, and DU, placed at a distance of 0.01 cm from a 20 MeV gamma source with the goal of determining the depth profile of the gamma

beam attenuation. Approximately 3 cm of DU will attenuate over 90% of the beam, while 4m of NAVA ALIGA and 6 cm of lead is required to achieve the same.

Based on these findings a new geometry was developed to understand the efficiency of the combination of DU and NAVA ALIGA as a shielding block. The thickness of DU the block was placed every 1 m, as shown in Fig. 7.1, for a total of six models. The surface of the block facing the source is covered with 3 cm of concrete (to cover the 3 cm layer of DU from the environment).

7.3 Most Optimal Combination of DU With NAVA ALIGA for Enhanced Shielding

In order to most accurately determine the optimal combination of concrete and DU, information about not only the shielding capabilities is needed but also cost of processing, shipment, and handling of DU. Due to the difficulty faced with obtaining DU for physical testing due to toxicity and licensing requirements, information with regard to cost and availability for shipping was not available [31]. Estimations for the most optimal combination of DU and concrete were made based solely off of the shielding capabilities as determined from MCNP6.

The simulations demonstrate that DU is a much more effective high energy gamma-rays shield than concrete but only marginally more effective than lead. This is shown in Fig. 7.2 where the flux is reduced to zero most quickly when the DU is placed 3 cm from the face of the wall closest to the source. Fig. 7.3 shows a semilog plot of the y-axis, where a sudden drop in flux can be seen wherever DU is placed in the shield. 3 cm of DU decreases the flux by approximately one order of magnitude, seen in Fig. 7.3. Concrete requires approximately 40 cm to decrease flux by the same amount.

The results of the simulations indicate that in order for DU to be most effective it is best to place it as close to the source as possible. DU is a much more effective shield of high energy gamma rays than concrete due to its higher density. In order to maximize the shielding capabilities of DU it is best to utilize it for the highest energy gamma-rays. This is done by placing DU at a shallow depth within the concrete. Depleted uranium is only marginally more effective at shielding high energy gamma-rays than lead.

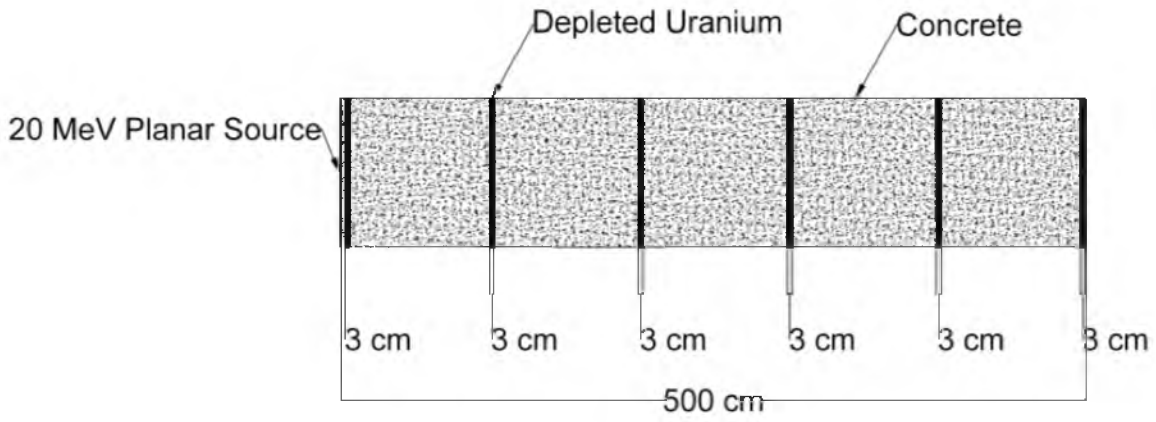

Fig. 7.1: MCNP6 model of DU and NAVA ALIGA. In every position 1 DU layer represents a model (in total six models).

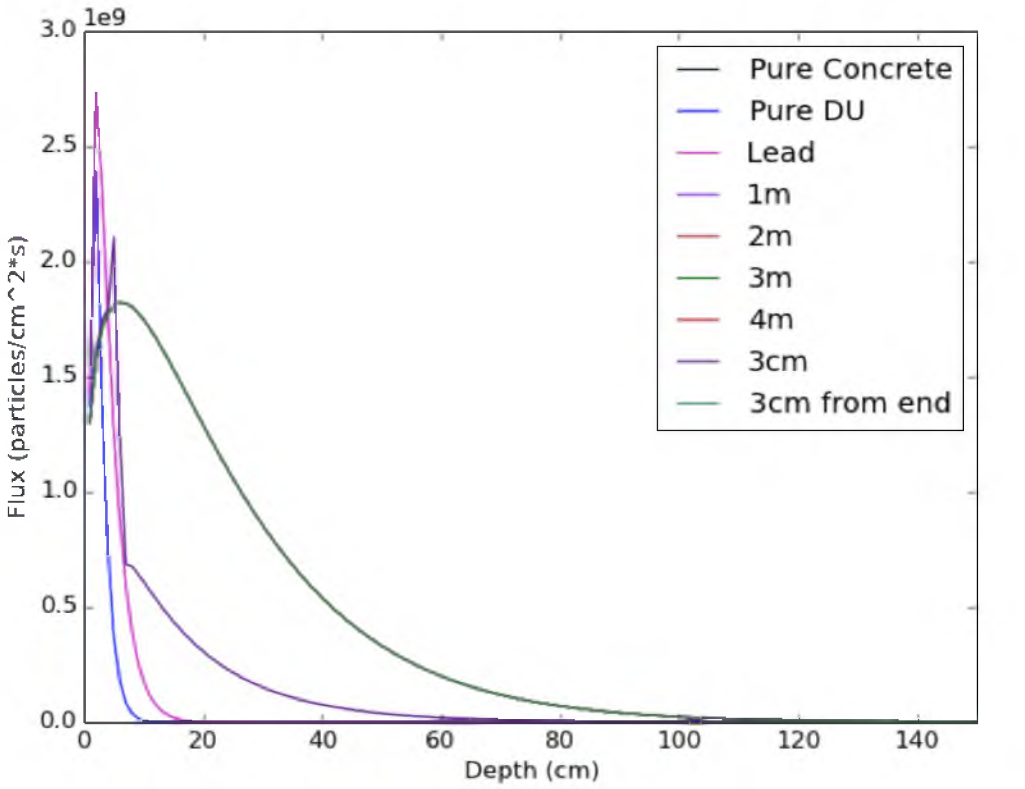

Fig. 7.2: The effect of NAVA ALIGA and DU on attenuation of 20 MeV gamma flux

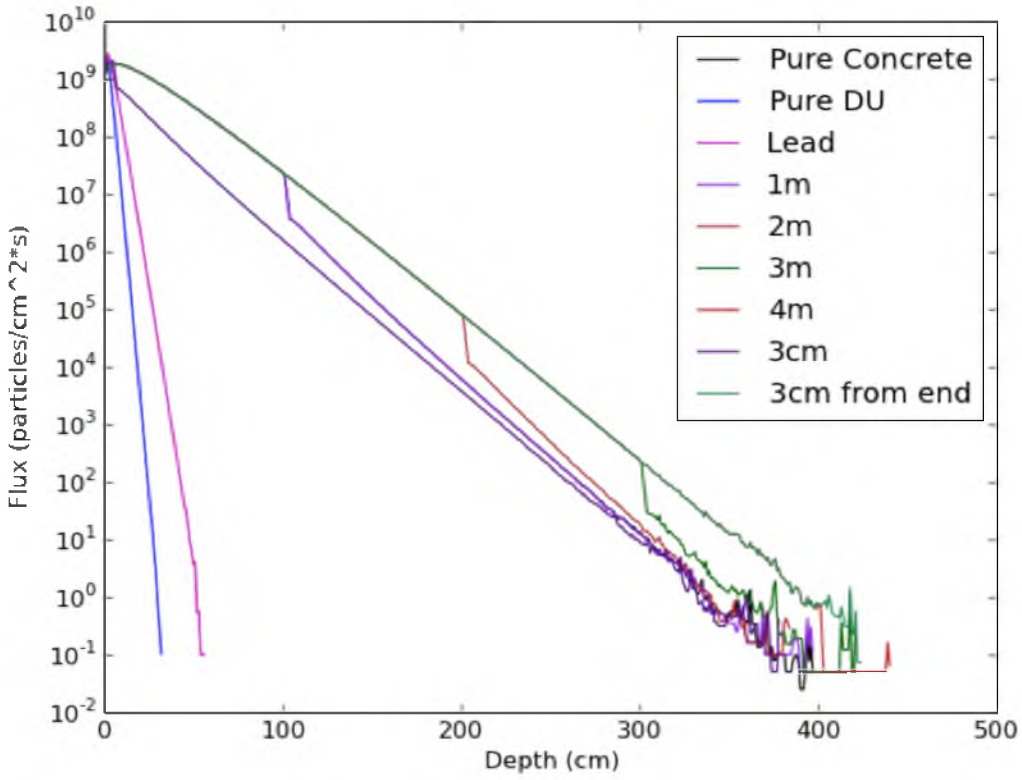

Fig. 7.3: Semilog depiction of the effect of NAVA ALIGA and DU on attenuation of 20 MeV gamma flux

CHAPTER 8

CONCLUSION AND FUTURE WORK

8.1 Conclusion

NAVA ALIGA was disclosed with The University of Utah in 2012 as the material of superior choice to provide long-term durable concrete with the same properties concrete currently used, but allows for the material to be recycled as a regular concrete. NAVA ALIGA will never be activated above the clearance level after long-term exposure to neutron irradiation. Using INAA, a definitive criterion has been established for identifying low-activation concrete materials. This criterion was tested on a number of cement and aggregate samples, and concrete materials were positively identified along the Wasatch Front that meets these specifications for low-activation concrete. Despite discrepancies in measurements between UNEF gamma spectrometry measurements, ISU measurements, and ICP-MS, all data falls below IAEA clearance levels, even in a worstcase scenario where the highest concentrations are deemed most correct.

It has been shown that using materials carefully selected for low-activation characteristics can be used to create high-strength concrete mixtures needed for structural applications. Using only the basic ingredients of water, cement, sand, and rock, a mixture of concrete was created with greater than 6000-psi compressive strength. Using the admixtures common in concrete production such as water-reducer and plasticizer as well as improved curing techniques, this strength will only increase. Despite the high permeability of the mixture, it still shows promise to be used in nuclear environments. The use of aforementioned admixtures will also greatly increase the density of the concrete, which is directly related to the permeability of the concrete.

With a greater density than lead, which is commonly used as a radiation shield, it was expected and shown through MCNP6 simulations that DU when combined with concrete does create an effective biological shield. Despite the effectiveness of DU at shielding radiation, the toxicity of it makes it difficult to perform real testing. Most of the nations' supply of DU is in the form of UF6 and not in metallic form [33]. When UF6 is mixed with water, highly toxic HF gas is created, which can create hydrofluoric acid upon contact with skin. For this reason all UF6 must be converted to some other form in order to be useful for shielding.

8.2 Future Work

Future work should include recalibration of all HPGe detectors housed within UNEF. This is expected to improve analysis of measured samples. After recalibration a larger number of samples used in NAVA ALIGA could be tested and compared against Idaho State University measurements. This would help to verify accuracy of measurements and establish greater confidence in all future analysis. Additionally, a greater number of concrete supplies should be tested. Cement in particular is composed of a number of different materials brought together through a crushing and heating process where trace element contamination could occur. These individual components should be tested at each stage of the cement manufacturing process to identify if contamination from cobalt occurs in hardened steel milling bits. If there is one particular stage of the manufacturing process where contamination is occurring, changes in the crushing or grinding could be made to create a broader range of materials suitable for low-activation concrete usage. These changes in the grinding process could also be implemented in coarse and fine aggregate milling.

Future research will utilize the methods of testing for low-activation characteristics to branch out from the Wasatch Front and identify concrete supplies suitable for low-activation where nuclear power plants are planned for construction. Concrete admixtures should also be tested to ensure that they also meet low-activation criteria before being used in concrete mixtures. As admixtures are qualified, they will create stronger, denser, and less permeable concrete more suitable for use in nuclear infrastructure including containment domes, spent fuel pools, and spent fuel storage casks. Alternative materials could also be tested such as pozzolan and geopolymer cement. These materials are known for their increased weather ability and capacity for higher strength.

The IAEA currently has low level and very low level waste standards; however, the United States currently does not have a very low level waste category. Work is currently being done to establish a very low level category for radioactive waste that most activated concrete would fall into. As a result, future analysis should be based on a change from LLRW to VLLW for the broadest applicability [1].

Concrete is also subject to contamination from not only activation but also liquid spills and gaseous diffusion. At the Connecticut Yankee and Trojan nuclear power plants, levels of diffused tritium and C-14 exceeded IAEA clearance levels by themselves without taking into account the activation of the concrete [1]. Admixtures that densify concrete might offer a solution for concrete to resist gaseous diffusion as well as chemical spills that reactor containment domes are subjected to.

APPENDIX A

CEMENT, COARSE, AND FINE AGGREGATE SPECTRA

The following figures represent the samples activated through INAA and then counted on station 8 in UNEF. All spectra and GENIE 2000 files are saved in the following location: C:\GENIE2K\CAMFILES\Steve Burnham\Sample Recounts 2\.

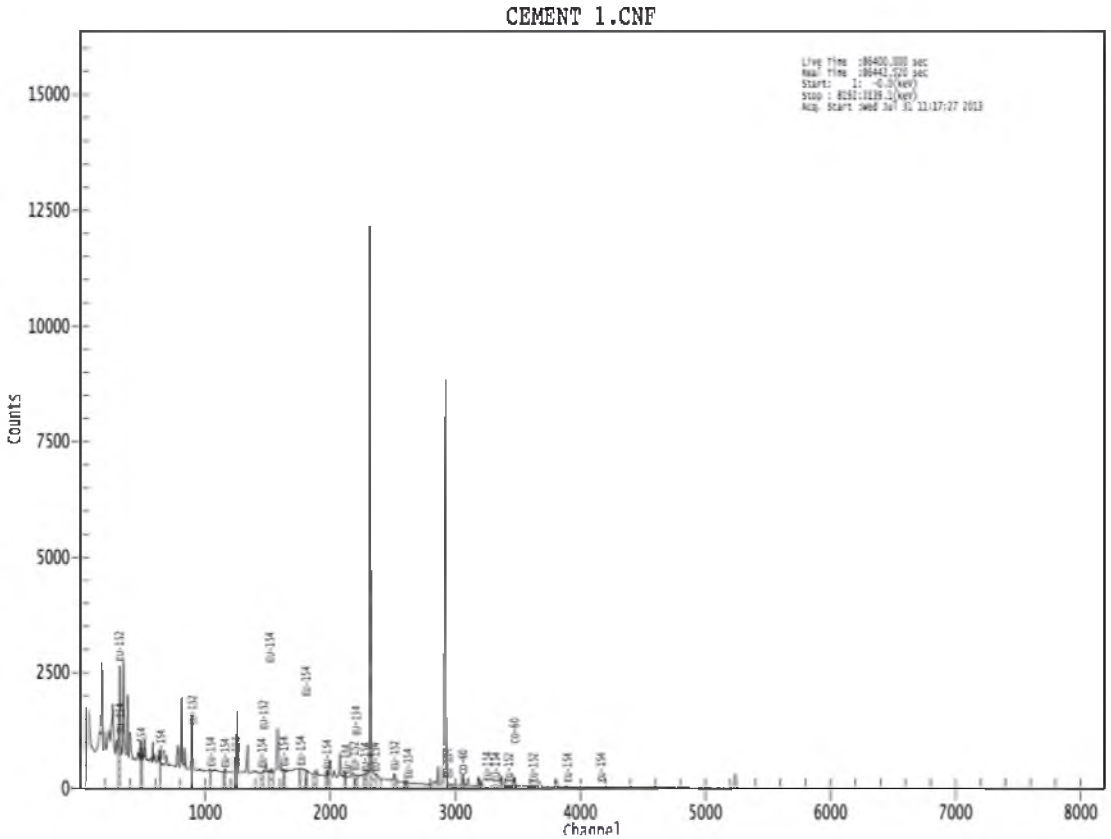

Fig. A.1: Gamma spectrum from cement 1

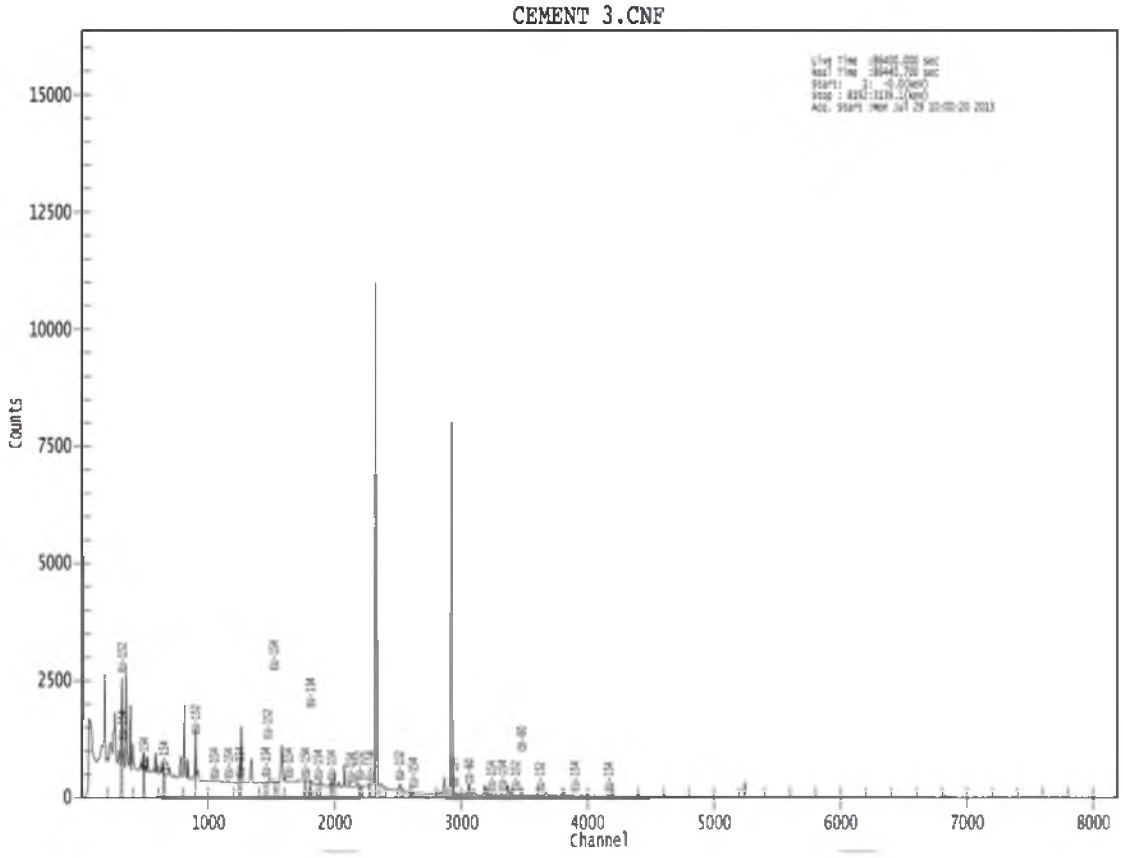

Fig. A.2: Gamma spectrum from cement 3

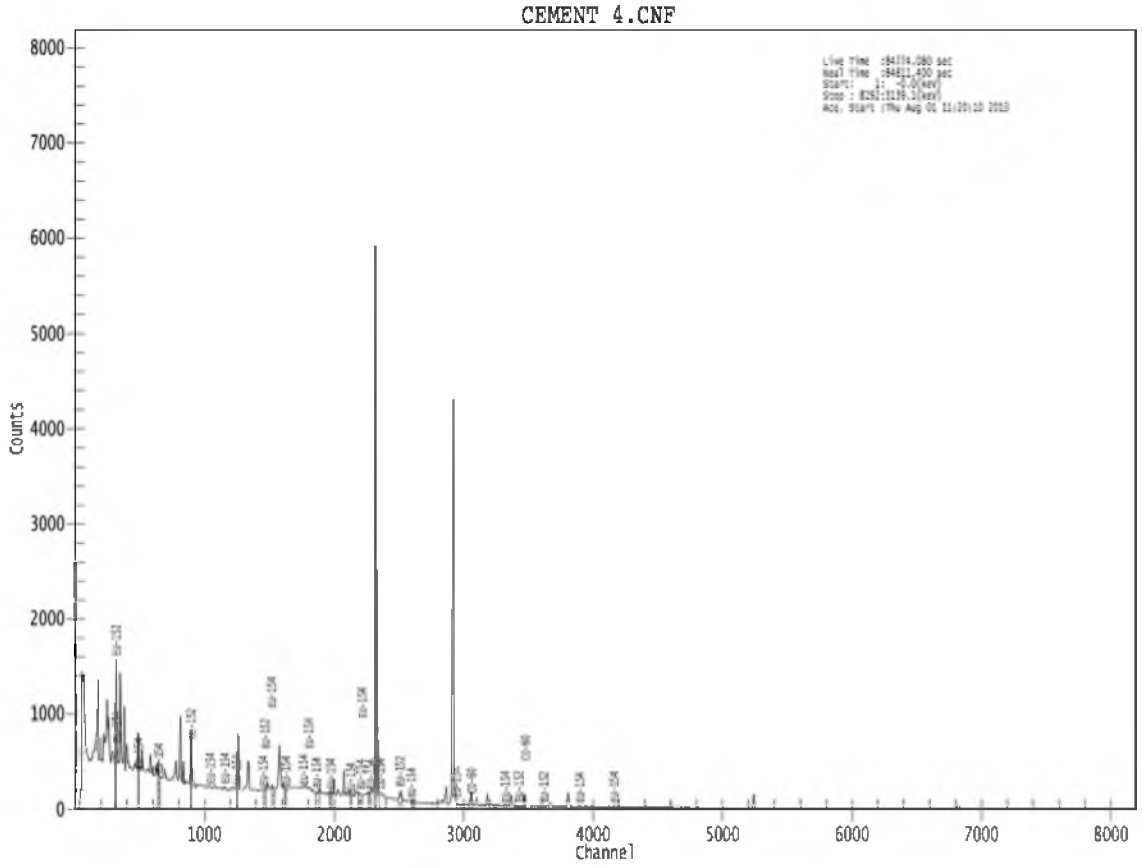

Fig. A.3: Gamma spectrum from cement 4

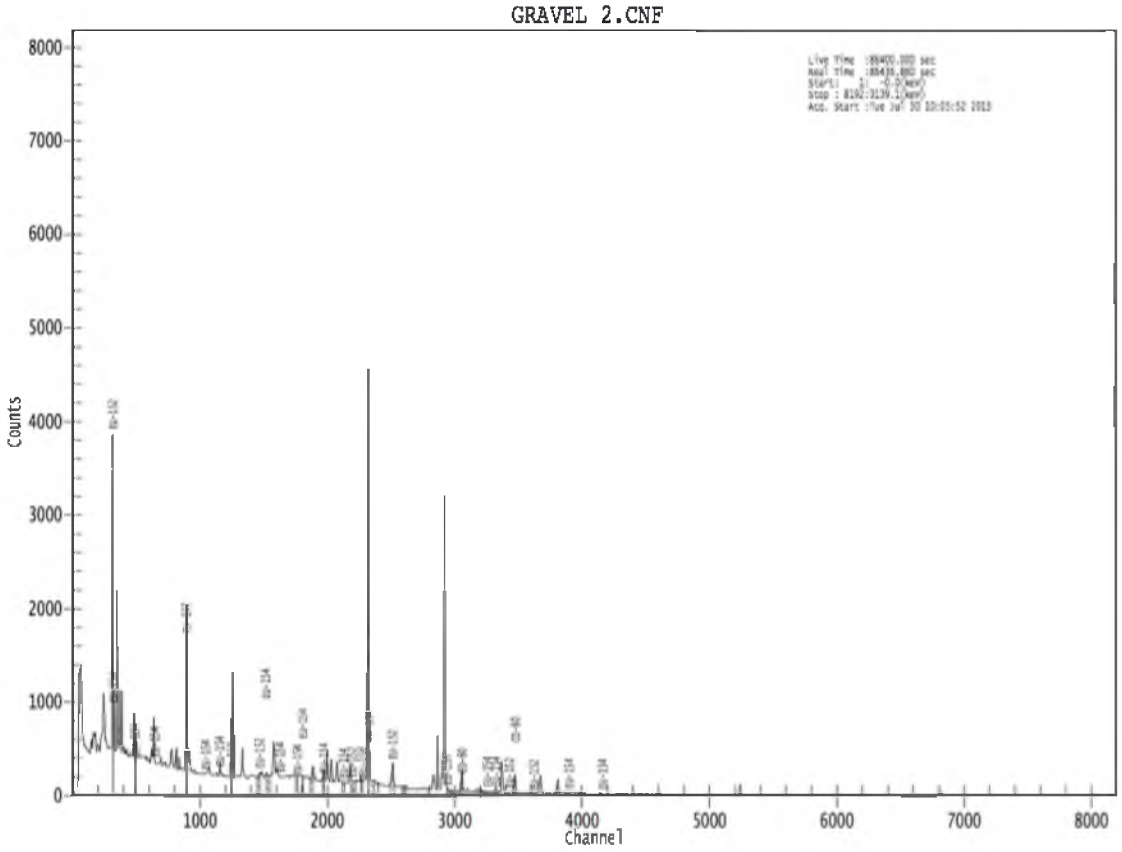

Fig. A.4: Gamma spectrum from gravel 2

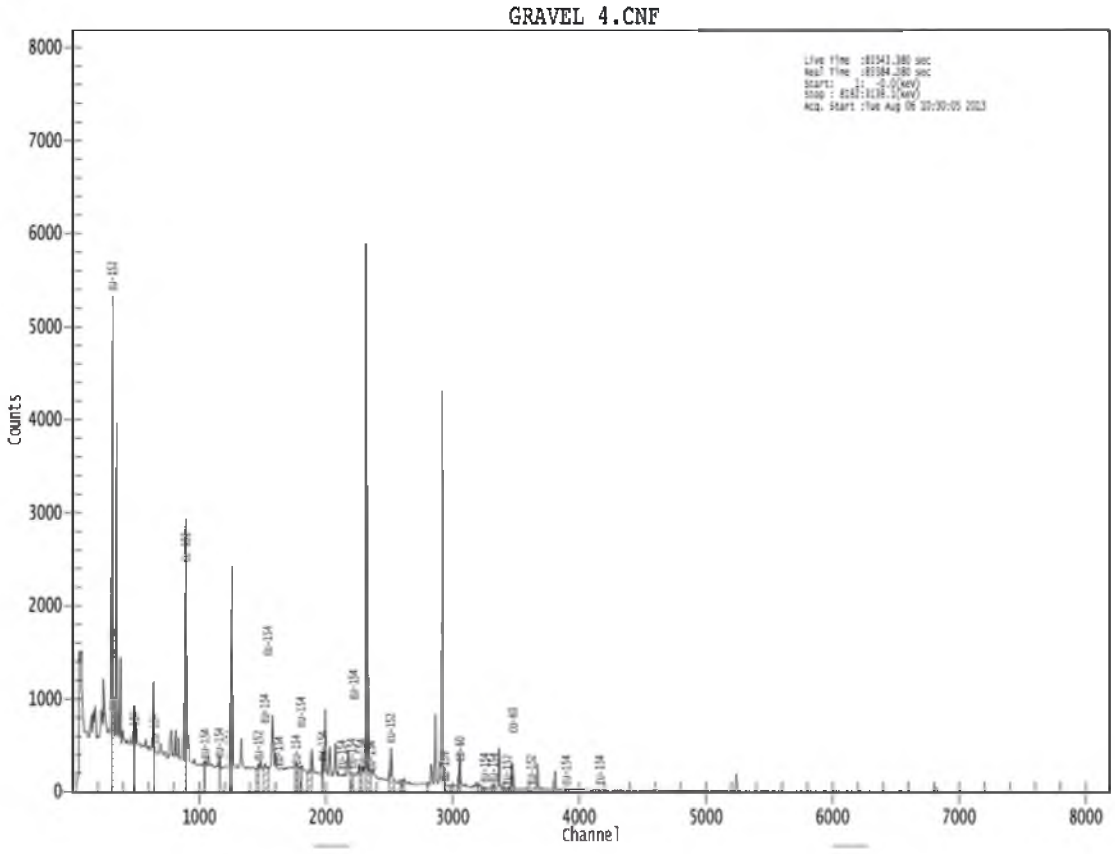

Fig. A.5: Gamma spectrum from gravel 4

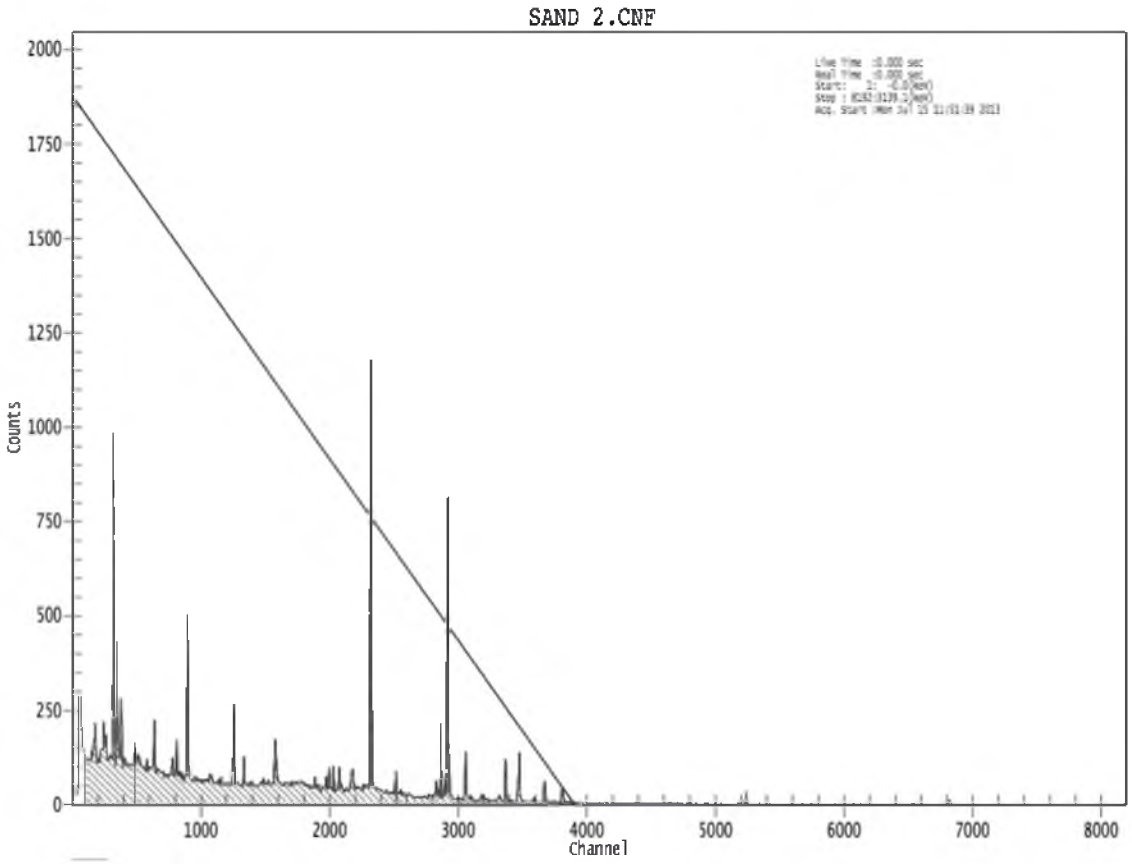

Fig. A.6: Gamma spectrum from sand 2

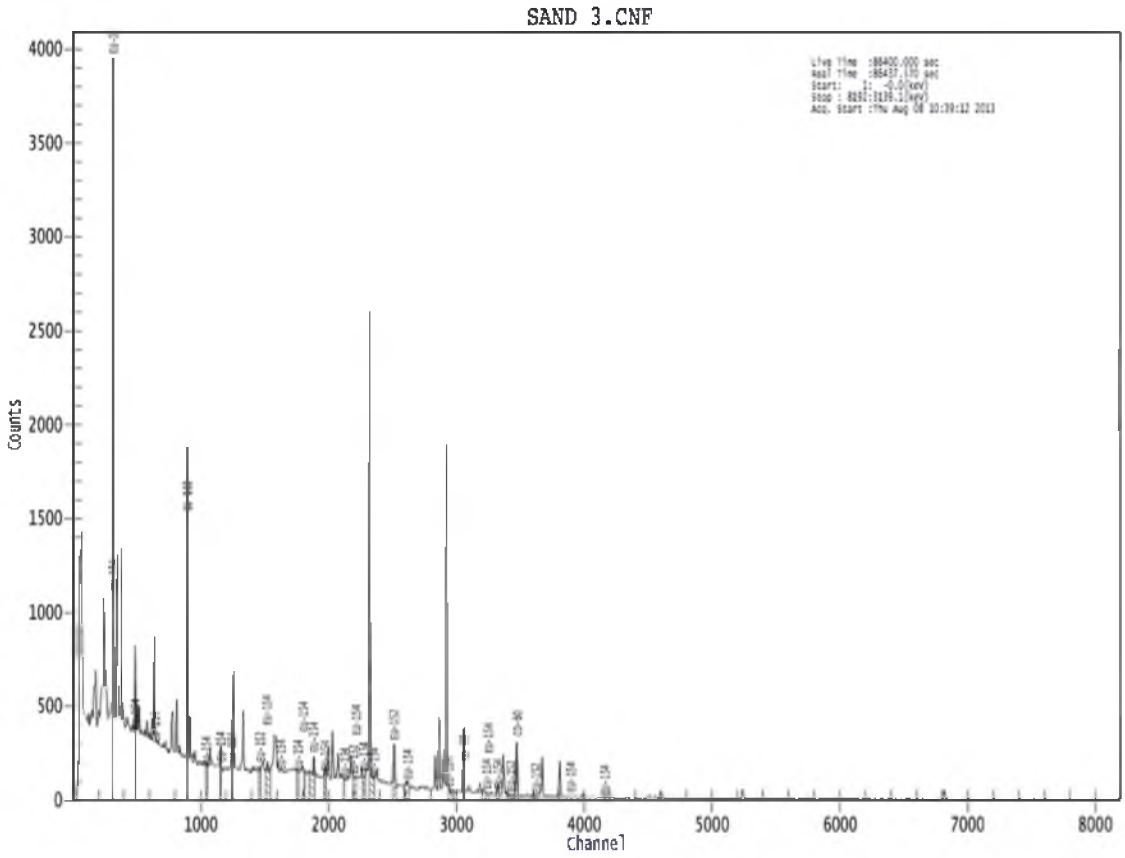

Fig. A.7: Gamma spectrum from Sand 3

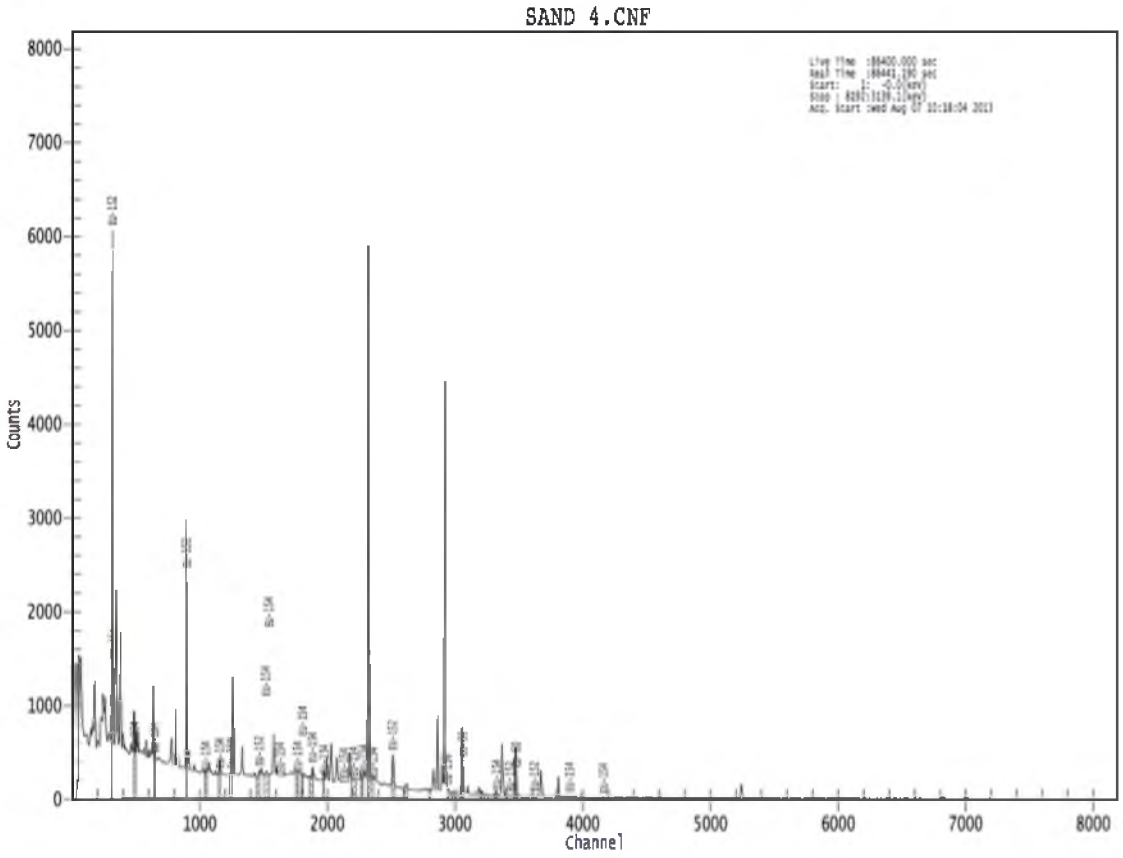

Fig. A.8: Gamma spectrum from sand 4

APPENDIX B

MCNP6 INPUT FILES FOR DU SHIELDING ANALYSIS

The following are the input files that were used to run three of the simulations used to determine the optimum combination and placement of DU with concrete for a biological shield. The first input file is for a 100% concrete wall. The second input file is for a 100% DU wall. The final input file is for a mixture of Concrete with 3 cm of DU placed at a depth from 3 cm to 7 cm.

Concrete Wall
C ********** BLOCK 1 – cells
1 10 -2.3 -101 102 702 -703 806 -807 IMP:N=1 IMP:P=1
2 10 -2.3 -102 103 702 -703 806 -807 IMP:N=1 IMP:P=1
3 10 -2.3 -103 104 702 -703 806 -807 IMP:N=1 IMP:P=1
4 10 -2.3 -104 105 702 -703 806 -807 IMP:N=1 IMP:P=1
5 10 -2.3 -105 106 702 -703 806 -807 IMP:N=1 IMP:P=1
6 10 -2.3 -106 107 702 -703 806 -807 IMP:N=1 IMP:P=1
7 10 -2.3 -107 108 702 -703 806 -807 IMP:N=1 IMP:P=1
8 10 -2.3 -108 109 702 -703 806 -807 IMP:N=1 IMP:P=1
9 10 -2.3 -109 110 702 -703 806 -807 IMP:N=1 IMP:P=1
10 10 -2.3 -110 111 702 -703 806 -807 IMP:N=1 IMP:P=1
11 10 -2.3 -111 112 702 -703 806 -807 IMP:N=1 IMP:P=1
12 10 -2.3 -112 113 702 -703 806 -807 IMP:N=1 IMP:P=1
13 10 -2.3 -113 114 702 -703 806 -807 IMP:N=1 IMP:P=1
14 10 -2.3 -114 115 702 -703 806 -807 IMP:N=1 IMP:P=1
15 10 -2.3 -115 116 702 -703 806 -807 IMP:N=1 IMP:P=1
16 10 -2.3 -116 117 702 -703 806 -807 IMP:N=1 IMP:P=1
17 10 -2.3 -117 118 702 -703 806 -807 IMP:N=1 IMP:P=1
18 10 -2.3 -118 119 702 -703 806 -807 IMP:N=1 IMP:P=1
19 10 -2.3 -119 120 702 -703 806 -807 IMP:N=1 IMP:P=1
20 10 -2.3 -120 121 702 -703 806 -807 IMP:N=1 IMP:P=1
21 10 -2.3 -121 122 702 -703 806 -807 IMP:N=1 IMP:P=1
22 10 -2.3 -122 123 702 -703 806 -807 IMP:N=1 IMP:P=1
23 10 -2.3 -123 124 702 -703 806 -807 IMP:N=1 IMP:P=1
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435 10 -2.3 -535 536 702 -703 806 -807 IMP:N=1 IMP:P=1
436 10 -2.3 -536 537 702 -703 806 -807 IMP:N=1 IMP:P=1
437 10 -2.3 -537 538 702 -703 806 -807 IMP:N=1 IMP:P=1
438 10 -2.3 -538 539 702 -703 806 -807 IMP:N=1 IMP:P=1
439 10 -2.3 -539 540 702 -703 806 -807 IMP:N=1 IMP:P=1
440 10 -2.3 -540 541 702 -703 806 -807 IMP:N=1 IMP:P=1
441 10 -2.3 -541 542 702 -703 806 -807 IMP:N=1 IMP:P=1
442 10 -2.3 -542 543 702 -703 806 -807 IMP:N=1 IMP:P=1
443 10 -2.3 -543 544 702 -703 806 -807 IMP:N=1 IMP:P=1
444 10 -2.3 -544 545 702 -703 806 -807 IMP:N=1 IMP:P=1
445 10 -2.3 -545 546 702 -703 806 -807 IMP:N=1 IMP:P=1
446 10 -2.3 -546 547 702 -703 806 -807 IMP:N=1 IMP:P=1
447 10 -2.3 -547 548 702 -703 806 -807 IMP:N=1 IMP:P=1
448 10 -2.3 -548 549 702 -703 806 -807 IMP:N=1 IMP:P=1

449 10 -2.3 -549 550 702 -703 806 -807 IMP:N=1 IMP:P=1
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452 10 -2.3 -552 553 702 -703 806 -807 IMP:N=1 IMP:P=1
453 10 -2.3 -553 554 702 -703 806 -807 IMP:N=1 IMP:P=1
454 10 -2.3 -554 555 702 -703 806 -807 IMP:N=1 IMP:P=1
455 10 -2.3 -555 556 702 -703 806 -807 IMP:N=1 IMP:P=1
456 10 -2.3 -556 557 702 -703 806 -807 IMP:N=1 IMP:P=1
457 10 -2.3 -557 558 702 -703 806 -807 IMP:N=1 IMP:P=1
458 10 -2.3 -558 559 702 -703 806 -807 IMP:N=1 IMP:P=1
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461 10 -2.3 -561 562 702 -703 806 -807 IMP:N=1 IMP:P=1
462 10 -2.3 -562 563 702 -703 806 -807 IMP:N=1 IMP:P=1
463 10 -2.3 -563 564 702 -703 806 -807 IMP:N=1 IMP:P=1
464 10 -2.3 -564 565 702 -703 806 -807 IMP:N=1 IMP:P=1
465 10 -2.3 -565 566 702 -703 806 -807 IMP:N=1 IMP:P=1
466 10 -2.3 -566 567 702 -703 806 -807 IMP:N=1 IMP:P=1
467 10 -2.3 -567 568 702 -703 806 -807 IMP:N=1 IMP:P=1
468 10 -2.3 -568 569 702 -703 806 -807 IMP:N=1 IMP:P=1
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470 10 -2.3 -570 571 702 -703 806 -807 IMP:N=1 IMP:P=1
471 10 -2.3 -571 572 702 -703 806 -807 IMP:N=1 IMP:P=1
472 10 -2.3 -572 573 702 -703 806 -807 IMP:N=1 IMP:P=1
473 10 -2.3 -573 574 702 -703 806 -807 IMP:N=1 IMP:P=1
474 10 -2.3 -574 575 702 -703 806 -807 IMP:N=1 IMP:P=1
475 10 -2.3 -575 576 702 -703 806 -807 IMP:N=1 IMP:P=1
476 10 -2.3 -576 577 702 -703 806 -807 IMP:N=1 IMP:P=1
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478 10 -2.3 -578 579 702 -703 806 -807 IMP:N=1 IMP:P=1
479 10 -2.3 -579 580 702 -703 806 -807 IMP:N=1 IMP:P=1
480 10 -2.3 -580 581 702 -703 806 -807 IMP:N=1 IMP:P=1
481 10 -2.3 -581 582 702 -703 806 -807 IMP:N=1 IMP:P=1
482 10 -2.3 -582 583 702 -703 806 -807 IMP:N=1 IMP:P=1
483 10 -2.3 -583 584 702 -703 806 -807 IMP:N=1 IMP:P=1
484 10 -2.3 -584 585 702 -703 806 -807 IMP:N=1 IMP:P=1
485 10 -2.3 -585 586 702 -703 806 -807 IMP:N=1 IMP:P=1
486 10 -2.3 -586 587 702 -703 806 -807 IMP:N=1 IMP:P=1
487 10 -2.3 -587 588 702 -703 806 -807 IMP:N=1 IMP:P=1
488 10 -2.3 -588 589 702 -703 806 -807 IMP:N=1 IMP:P=1
489 10 -2.3 -589 590 702 -703 806 -807 IMP:N=1 IMP:P=1
490 10 -2.3 -590 591 702 -703 806 -807 IMP:N=1 IMP:P=1
491 10 -2.3 -591 592 702 -703 806 -807 IMP:N=1 IMP:P=1
492 10 -2.3 -592 593 702 -703 806 -807 IMP:N=1 IMP:P=1
493 10 -2.3 -593 594 702 -703 806 -807 IMP:N=1 IMP:P=1
494 10 -2.3 -594 595 702 -703 806 -807 IMP:N=1 IMP:P=1
495 10 -2.3 -595 596 702 -703 806 -807 IMP:N=1 IMP:P=1
496 10 -2.3 -596 597 702 -703 806 -807 IMP:N=1 IMP:P=1
497 10 -2.3 -597 598 702 -703 806 -807 IMP:N=1 IMP:P=1
498 10 -2.3 -598 599 702 -703 806 -807 IMP:N=1 IMP:P=1

```
499 10 -2.3 -599 600 702 -703 806 -807 IMP:N=1 IMP:P=1
500 10 -2.3 -600 100 702 -703 806 -807 IMP:N=1 IMP:P=1
3333 0 -908 (-100:101:-702:703:-806:807) IMP:N=1 IMP:P=1
4444 0 908 -909 IMP:N=1 IMP:P=1
5555 0 909 IMP:N=0 IMP:P=0
C ********** BLOCK 2 – surfaces/macrobodies
100 px -250
101 px 250
102 px 249
103 px 248
104 px 247
105 px 246
106 px 245
107 px 244
108 px 243
109 px 242
110 px 241
111 px 240
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113 px 238
114 px 237
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119 px 232
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127 px 224
128 px 223
129 px 222
130 px 221
131 px 220
132 px 219
133 px 218
134 px 217
135 px 216
136 px 215
137 px 214
138 px 213
139 px 212
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141 px 210
142 px 209
143 px 208
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144 px 207
145 px 206
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147 px 204
148 px 203
149 px 202
150 px 201
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152 px 199
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244 px 107
245 px 106
246 px 105
247 px 104
248 px 103
249 px 102
250 px 101
251 px 100
252 px 99
253 px 98
254 px 97
255 px 96
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257 px 94
258 px 93
259 px 92
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261 px 90
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333 px 18
334 px 17
335 px 16
336 px 15
337 px 14
338 px 13
339 px 12
340 px 11
341 px 10
342 px 9
343 px 8

344 px 7
345 px 6
346 px 5
347 px 4
348 px 3
349 px 2
350 px 1
351 px 0
352 px -1
353 px -2
354 px -3
355 px -4
356 px -5
357 px -6
358 px -7
359 px -8
360 px -9
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591 px -240
592 px -241
593 px -242

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594 px -243
595 px -244
596 px -245
597 px -246
598 px -247
599 px -248
600 px -249
702 py -50
703 py 50
806 pz -50
807 pz 50
908 so 500
909 so 501
C ***************** BLOCK 3 – data
SDEF POS=364 0 0 x=364 y=d1 z=d2 par=2 ERG=20
AXS= 1 0 0 Vec= 1 0 0 DIR=-1
SI1 -50 50
SP1 0 1
SI2 -50 50
SP2 0 1
mode n p
c ---------------------------------
c CONCRETE MATERIAL
c ---------------------------------
c m1 21045.70c -0.0000017132
c 26058.66c -0.0040468037
c 27059.66c -0.0000001793
c 34074.70c -0.0000039443
c 38084.70c -0.0005198729
c 44102.70c -0.0000008229
c 51121.70c -0.0000009688
c 51123.70c -0.0000008072
c 56130.70c -0.0001308765
c 55133.70c -0.0000007725
c 57139.70c -0.0000088666
c 58140.70c -0.0000240198
c 62152.70c -0.0000010743
c 63151.66c -0.0000000501
c 64153.70c -0.0000046425
c 65159.70c -0.0000002731
c 72180.70c -0.0000000607
c 1001.66c -0.006160624
c 6000.66c -0.174368545
c 8016.66c -0.408253294
c 11023.66c -0.000268719
c 12000.66c -0.032495051
c 13027.66c -0.010778604
c 14000.60c -0.034316367
c 19000.66c -0.00113459
c 20000.66c -0.319775191
```

```
c 26000.50c -0.007743078
m2 92238.70c -1
m10 1001.66c -0.00619 $ Concrete
6000.66c -0.17520
8016.66c -0.41020
11023.66c -0.00027
12000.66c -0.03265
13027.66c -0.01083
14000.60c -0.03448
19000.66c -0.00114
20000.66c -0.32130
26000.50c -0.00778
c m11 7014.66c 0.0000381259 $Air
c 8016.66c 0.0000095012
c 18000.59c 0.0000001664
c act nonfiss = p dg = lines
c F74:p 3
c E74 1e-6 8192i 3
c T74 1e6 1e18 T
F2:p 101
F12:p 105
F22:p 109
F32:p 113
F42:p 117
F52:p 121
F62:p 125
F72:p 129
F82:p 133
F92:p 137
F102:p 141
F112:p 145
F122:p 149
F132:p 153
F142:p 157
F152:p 161
F162:p 165
F172:p 169
F182:p 173
F192:p 177
F202:p 181
F212:p 185
F222:p 189
F232:p 193
F242:p 197
F252:p 201
F262:p 205
F272:p 209
F282:p 213
F292:p 217
F302:p 221
```

F312:p 225
F322:p 229
F332:p 233
F342:p 237
F352:p 241
F362:p 245
F372:p 249
F382:p 253
F392:p 257
F402:p 261
F412:p 265
F422:p 269
F432:p 273
F442:p 277
F452:p 281
F462:p 285
F472:p 289
F482:p 293
F492:p 297
F502:p 301
F512:p 305
F522:p 309
F532:p 313
F542:p 317
F552:p 321
F562:p 325
F572:p 329
F582:p 333
F592:p 337
F602:p 341
F612:p 345
F622:p 349
F632:p 353
F642:p 357
F652:p 361
F662:p 365
F672:p 369
F682:p 373
F692:p 377
F702:p 381
F712:p 385
F722:p 389
F732:p 393
F742:p 397
F752:p 401
F762:p 405
F772:p 409
F782:p 413
F792:p 417
F802:p 421

F812:p 425
F822:p 429
F832:p 433
F842:p 437
F852:p 441
F862:p 445
F872:p 449
F882:p 453
F892:p 457
F902:p 461
F912:p 465
F922:p 469
F932:p 473
F942:p 477
F952:p 481
F962:p 485
F972:p 489
F982:p 493
F992:p 497
nps 40e9

Depleted Uranium Wall
C ********** BLOCK 1 – cells
1 2 -19.1 -101 102 702 -703 806 -807 IMP:N=1 IMP:P=1
2 2 -19.1 -102 103 702 -703 806 -807 IMP:N=1 IMP:P=1
3 2 -19.1 -103 104 702 -703 806 -807 IMP:N=1 IMP:P=1
4 2 -19.1 -104 105 702 -703 806 -807 IMP:N=1 IMP:P=1
5 2 -19.1 -105 106 702 -703 806 -807 IMP:N=1 IMP:P=1
6 2 -19.1 -106 107 702 -703 806 -807 IMP:N=1 IMP:P=1
7 2 -19.1 -107 108 702 -703 806 -807 IMP:N=1 IMP:P=1
8 2 -19.1 -108 109 702 -703 806 -807 IMP:N=1 IMP:P=1
9 2 -19.1 -109 110 702 -703 806 -807 IMP:N=1 IMP:P=1
10 2 -19.1 -110 111 702 -703 806 -807 IMP:N=1 IMP:P=1
11 2 -19.1 -111 112 702 -703 806 -807 IMP:N=1 IMP:P=1
12 2 -19.1 -112 113 702 -703 806 -807 IMP:N=1 IMP:P=1
13 2 -19.1 -113 114 702 -703 806 -807 IMP:N=1 IMP:P=1
14 2 -19.1 -114 115 702 -703 806 -807 IMP:N=1 IMP:P=1
15 2 -19.1 -115 116 702 -703 806 -807 IMP:N=1 IMP:P=1
16 2 -19.1 -116 117 702 -703 806 -807 IMP:N=1 IMP:P=1
17 2 -19.1 -117 118 702 -703 806 -807 IMP:N=1 IMP:P=1
18 2 -19.1 -118 119 702 -703 806 -807 IMP:N=1 IMP:P=1
19 2 -19.1 -119 120 702 -703 806 -807 IMP:N=1 IMP:P=1
20 2 -19.1 -120 121 702 -703 806 -807 IMP:N=1 IMP:P=1
21 2 -19.1 -121 122 702 -703 806 -807 IMP:N=1 IMP:P=1
22 2 -19.1 -122 123 702 -703 806 -807 IMP:N=1 IMP:P=1
23 2 -19.1 -123 124 702 -703 806 -807 IMP:N=1 IMP:P=1
24 2 -19.1 -124 125 702 -703 806 -807 IMP:N=1 IMP:P=1
25 2 -19.1 -125 126 702 -703 806 -807 IMP:N=1 IMP:P=1
26 2 -19.1 -126 127 702 -703 806 -807 IMP:N=1 IMP:P=1
27 2 -19.1 -127 128 702 -703 806 -807 IMP:N=1 IMP:P=1

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28 2 -19.1 -128 129 702 -703 806 -807 IMP:N=1 IMP:P=1
29 2 -19.1 -129 130 702 -703 806 -807 IMP:N=1 IMP:P=1
30 2 -19.1 -130 131 702 -703 806 -807 IMP:N=1 IMP:P=1
31 2 -19.1 -131 132 702 -703 806 -807 IMP:N=1 IMP:P=1
32 2 -19.1 -132 133 702 -703 806 -807 IMP:N=1 IMP:P=1
33 2 -19.1 -133 134 702 -703 806 -807 IMP:N=1 IMP:P=1
34 2 -19.1 -134 135 702 -703 806 -807 IMP:N=1 IMP:P=1
35 2 -19.1 -135 136 702 -703 806 -807 IMP:N=1 IMP:P=1
36 2 -19.1 -136 137 702 -703 806 -807 IMP:N=1 IMP:P=1
37 2 -19.1 -137 138 702 -703 806 -807 IMP:N=1 IMP:P=1
38 2 -19.1 -138 139 702 -703 806 -807 IMP:N=1 IMP:P=1
39 2 -19.1 -139 140 702 -703 806 -807 IMP:N=1 IMP:P=1
40 2 -19.1 -140 141 702 -703 806 -807 IMP:N=1 IMP:P=1
41 2 -19.1 -141 142 702 -703 806 -807 IMP:N=1 IMP:P=1
42 2 -19.1 -142 143 702 -703 806 -807 IMP:N=1 IMP:P=1
43 2 -19.1 -143 144 702 -703 806 -807 IMP:N=1 IMP:P=1
44 2 -19.1 -144 145 702 -703 806 -807 IMP:N=1 IMP:P=1
45 2 -19.1 -145 146 702 -703 806 -807 IMP:N=1 IMP:P=1
46 2 -19.1 -146 147 702 -703 806 -807 IMP:N=1 IMP:P=1
47 2 -19.1 -147 148 702 -703 806 -807 IMP:N=1 IMP:P=1
48 2 -19.1 -148 149 702 -703 806 -807 IMP:N=1 IMP:P=1
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377 2 -19.1 -477 478 702 -703 806 -807 IMP:N=1 IMP:P=1

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498 2 -19.1 -598 599 702 -703 806 -807 IMP:N=1 IMP:P=1
499 2 -19.1 -599 600 702 -703 806 -807 IMP:N=1 IMP:P=1
500 2 -19.1 -600 100 702 -703 806 -807 IMP:N=1 IMP:P=1
3333 0 -908 (-100:101:-702:703:-806:807) IMP:N=1 IMP:P=1
4444 0 908 -909 IMP:N=1 IMP:P=1
5555 0 909 IMP:N=0 IMP:P=0
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103 px 248
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107 px 244
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112 px 239
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123 px 228
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595 px -244
596 px -245
597 px -246
598 px -247
599 px -248
600 px -249
702 py -50
703 py 50
806 pz -50
807 pz 50
908 so 500
909 so 501
C ***************** BLOCK 3 – data
PRDMP ndmp=1
SDEF POS=251 0 0 x=251 y=d1 z=d2 par=2 ERG=20
AXS= 1 0 0 Vec= 1 0 0 DIR=-1
SI1 -50 50
SP1 0 1
SI2 -50 50
SP2 0 1
mode n p
c ———————————————————
c CONCRETE MATERIAL
c ———————————————————
c m1 21045.70c -0.0000017132
c 26058.66c -0.0040468037
c 27059.66c -0.0000001793
c 34074.70c -0.0000039443
```

```
c 38084.70c -0.0005198729
c 44102.70c -0.0000008229
c 51121.70c -0.0000009688
c 51123.70c -0.0000008072
c 56130.70c -0.0001308765
c 55133.70c -0.0000007725
c 57139.70c -0.0000088666
c 58140.70c -0.0000240198
c 62152.70c -0.0000010743
c 63151.66c -0.0000000501
c 64153.70c -0.0000046425
c 65159.70c -0.0000002731
c 72180.70c -0.0000000607
c 1001.66c -0.006160624
c 6000.66c -0.174368545
c 8016.66c -0.408253294
c 11023.66c -0.000268719
c 12000.66c -0.032495051
c 13027.66c -0.010778604
c 14000.60c -0.034316367
c 19000.66c -0.00113459
c 20000.66c -0.319775191
c 26000.50c -0.007743078
m2 92238.70c -1
m10 1001.66c -0.00619 $ Concrete
6000.66c -0.1752
8016.66c -0.4102
11023.66c -0.00027
12000.66c -0.03265
13027.66c -0.01083
14000.60c -0.03448
19000.66c -0.00114
20000.66c -0.16065
26000.50c -0.00778
92238.70c -0.16065
c m11 7014.66c 0.0000381259 $Air
c 8016.66c 0.0000095012
c 18000.59c 0.0000001664
c act nonfiss = p dg = lines
c F74:p 3
c E74 1e-6 8192i 3
c T74 1e6 1e18 T
F2:p 101
102
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nps 40e9

Concrete Wall 4cm to 7cm DU
C ********** BLOCK 1 – cells
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2 10 -2.3 -102 103 702 -703 806 -807 IMP:N=1 IMP:P=1
3 10 -2.3 -103 104 702 -703 806 -807 IMP:N=1 IMP:P=1
4 2 -19.1 -104 105 702 -703 806 -807 IMP:N=1 IMP:P=1

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348 px 3
349 px 2

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351 px 0
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600 px -249
702 py -50
703 py 50
806 pz -50
807 pz 50
908 so 500
909 so 501
C ********** BLOCK 3 – data
PRDMP ndmp=1
SDEF POS=251 0 0 x=251 y=d1 z=d2 par=2 ERG=20
AXS= 1 0 0 Vec= 1 0 0 DIR=-1
SI1 -50 50
SP1 0 1
SI2 -50 50
SP2 0 1
mode n p
c ---------------------------------
c CONCRETE MATERIAL
c ---------------------------------
c m1 21045.70c -0.0000017132
c 26058.66c -0.0040468037
c 27059.66c -0.0000001793
c 34074.70c -0.0000039443
c 38084.70c -0.0005198729
c 44102.70c -0.0000008229
c 51121.70c -0.0000009688
c 51123.70c -0.0000008072
c 56130.70c -0.0001308765
c 55133.70c -0.0000007725
c 57139.70c -0.0000088666
c 58140.70c -0.0000240198
c 62152.70c -0.0000010743
c 63151.66c -0.0000000501
c 64153.70c -0.0000046425
c 65159.70c -0.0000002731
c 72180.70c -0.0000000607
c 1001.66c -0.006160624
c 6000.66c -0.174368545
c 8016.66c -0.408253294
c 11023.66c -0.000268719
c 12000.66c -0.032495051
c 13027.66c -0.010778604
c 14000.60c -0.034316367
c 19000.66c -0.00113459
c 20000.66c -0.319775191
c 26000.50c -0.007743078
m2 92238.70c -1
m10 1001.66c -0.00619 $ Concrete
6000.66c -0.1752
8016.66c -0.4102
```

```
11023.66c -0.00027
12000.66c -0.03265
13027.66c -0.01083
14000.60c -0.03448
19000.66c -0.00114
20000.66c -0.16065
26000.50c -0.00778
92238.70c -0.16065
c m11 7014.66c 0.0000381259 $Air
c 8016.66c 0.0000095012
c 18000.59c 0.0000001664
c act nonfiss = p dg = lines
c F74:p 3
c E74 1e-6 8192i 3
c T74 1e6 1e18 T
F2:p 101
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